APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-482/91-13 Operating License No.: NPF-42

Docket: 50-482

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

Facility Name: Wolf Creek Generating Station (WCGS)

Inspection At: WCGS, Coffey County, Burlington, Kansas

Inspection Conducted: May 16 through June 26, 1991

Inspectors: L. L. Gundrum, Resident Inspector

W. B. Jones, Senior Project Engineer

Approved: A. T. Howell 7-29-91
A. T. Howell, Chief, Project Section D Date
Division of Reactor Projects

Inspection Summary

Inspection Conducted May 16 through June 26, 1991 (Report 50-482/91-13)

Areas Inspected: Routine, unannounced inspection including plant status; licensee event report followup; followup of previously identified NRC items; operational safety verification; surveillance observation; monthly maintenance observation; and preparation for refueling.

Results: During this inspection period, two violations were identified (failure to lock a valve in accordance with procedure, paragraph 3.2, and failure to comply with fire protection program requirements, paragraphs 4.1 and 4.2).

One unresolved item on the medical examinations for licensed personnel is identified in paragraph 4.3.

Strengths were noted in the performance of preventive maintenance activities. Related components were worked concurrently, which reduced the time safety-related and nonsafety-related equipment was inoperable. Good communications were noted between operations and results engineering personnel during the performance of the emergency diesel generator (EDG) room temperature profile test (paragraph 5.2 and Section 6).

Several weaknesses were noted in the surveillance program area. These weaknesses included decreasing the surveillance frequency for fire detection instruments without an adequate review or testing (paragraph 4.1); the failure to identify the Technical Specification requirement for performing leak rate testing on the containment emergency access hatch after entry (paragraph 4.5); and a long-standing EDG surveillance procedure weakness, which resulted in a delay in restoring an EDG to operable status (paragraph 5.1).

The licensee identified that personnel were not provided with sufficient technical information to adequately develop an inservice test procedure (paragraph 3.1). The inspectors noted a lack of administrative controls to require timely operability determinations (paragraph 4.1).

The inspectors identified that two quality assurance surveillances in the area of control of locked components failed to ensure that previous corrective actions in this area had been effective (paragraph 3.2). Concerns continued in the area of programmatic deficiency reporting because of delays noted in entering programmatic deficiency reports into the formal tracking system (paragraphs 4.1 and 7).

Acronyms and initialisms used in this report are provided in the attachment.

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
*G. D. Boyer, Director, Plant Operations
*O. L. Maynard, Deputy Director, Plant Operations
S. D. Austin, Shift Supervisor
*R. S. Benedict, Manager, Quality Control
*J. L. Blackwell, Fire Protection Specialist
*H. K. Chernoff, Supervisor, Licensing
A. B. Clason, Supervisor, Maintenance Engineering
*T. F. Deddens, Jr., Outage Manager
*M. E. Dingler, Manager, Nuclear Plant Engineering (NPE) Systems
*R. B. Flannigan, Manager, Nuclear Safety Engineering
*C. W. Fowler, Manager, Instrumentation & Control (I&C)
*R. L. Gourley, Supervisor, Mechanical Maintenance
*R. W. Holloway, Manager, Maintenance and Modifications
*R. K. Lewis, Supervisor, Results Engineering
*W. M. Lindsay, Manager, Quality Assurance (QA)
*R. L. Logsdon, Manager, Chemistry
*B. T. McKinney, Manager, Training
*T. S. Morrill, Manager, Radiation Protection
*D. G. Moseby, Supervisor, Operations
*W. B. Norton, Manager, Technical Support
*C. E. Parry, Director, Quality
*A. L. Payne, Manager, Supplier/Material, & Quality
J. M. Pippin, Director, NPE
*C. E. Rich, Jr., Supervisor, Electrical Maintenance
*C. M. Sprout, Section Manager, NPE, WCGS
J. D. Weeks, Manager, Operations
*S. G. Wideman, Senior Licensing Specialist
*M. G. Williams, Manager, Plant Support

The inspectors also contacted other members of the licensee's staff during the inspection period to discuss identified issues.

*Denotes those personnel in attendance at the exit meeting held on July 1, 1991.

2. PLANT STATUS

At the beginning of the inspection period, the plant was operating at 100 percent power. Reactor power was reduced to 60 percent on May 18, 1991, to conserve fuel. Reactor power was returned to 100 percent on May 31, 1991. The plant operated at or near 100 percent for the remainder of the inspection period. The planned power reduction during the past few months resulted in a savings of 36.1 effective full-power days. As of the end of the inspection period, the plant has operated for 401 consecutive days.

3. FOLLOWUP ON LICENSEE EVENT REPORTS (LER) AND PREVIOUSLY IDENTIFIED NRC INSPECTION FINDINGS

3.1 LER Followup (92700)

(Closed) LER 87-20: Engineered Safety Features Actuation - Control Room Ventilation Signal Caused by Paper Tape Breaking on Chlorine Monitor

Prior to the refueling outage conducted in 1988, the licensee had experienced numerous control room ventilation isolation system actuations. Many of the actuations resulted from malfunctions associated with the chlorine monitors. During the 1988 refueling outage, the licensee replaced the chlorine monitors in the control room ventilation systems. The new monitors have not experienced problems with spurious actuations. This LER is closed.

(Closed) LER 91-008: Engineered Safety Features Actuations Caused by Insufficient Self-Checking by Operator Who Erroneously Restored Radiation Monitor During Surveillance Testing

On May 28, 1991, Containment Purge Radiation Monitor GT RE-22 was placed in bypass to change the filter and to perform surveillance testing. Subsequently, fuel building exhaust radiation monitor GG RE-27, and the control room air intake monitor GK RE-05, were placed in bypass to change filters. Following the completion of filter changes, a licensed operator was instructed to restore GG RE-27 and GK RE-05. The operator restored GK RE-05 and inadvertently restored GT RE-22. I&C personnel, unaware that the monitor that they were working on had been restored, removed power from GT RE-22, which resulted in a containment purge isolation and a control room ventilation isolation. On receipt of these isolation signals, the control room operator immediately identified the cause, returned GT RE-22 to bypass, and restored the affected systems to their normal configuration. All equipment actuated as designed. The inspector has observed operations personnel placing radiation monitors in bypass and removing them from bypass on several occasions since this event. No problems were noted. The licensee's corrective action was to issue a letter to all operations personnel from the Manager, Operations, reinforcing the practice of self-checking. This corrective action was appropriate because it appears that this was an isolated incident. This LER is closed.

(Open) LER 91-007: Technical Specification (TS) Violation - Inadequate Testing of Component Cooling Water (CCW) to Reactor Coolant Pump Thermal Barrier Check Valves

On May 22, 1991, while determining the impact of a proposed design change on Procedure STS EG-206, Revision 0, "Component Cooling Water System Inservice Check Valve Test," the inservice testing engineer discovered that STS EG-206 does not adequately test CCW to reactor coolant pump thermal barrier Check Valves BB V0122, -V0152, -V0182, and -V0212 in their closed position. The test involves applying pressure downstream of the check valves while a pressure gauge upstream of the valve is used to determine whether system fluid is leaking by the valves. The inadequacy of this test method results from the presence of unisolated flowpaths upstream of the check valves; therefore, a pressure increase would not be detected and a failed check valve would not be identified.

The licensee's root cause determination as stated in the LER is that this test deficiency resulted from personnel not providing sufficient technical content during the initial development of the inservice test procedure. This determination also states that the test methodology deficiency has persisted through subsequent procedure reviews. No root cause was given as to why this deficiency was not detected during the 2-year procedure review process. This LER will remain open to further evaluate the adequacy of the root cause determination and the corrective actions taken.

(Closed) LER 90-002: Lack of Design Criteria For Actuation of Fire Suppression System

On March 14, 1990, the licensee identified that a Halon release in either engineered safety features (ESF) switchgear rooms would trip both Class 1E electrical equipment air conditioning units. A design oversight, involving installation criteria for the Halon release auxiliary shutdown relay, would have shut down both ESF ventilation trains. This design did not meet the single failure design criteria. The Halon release actuation circuitry was subsequently disabled and a fire watch established for both ESF switchgear rooms.

Plant Modification Request 03283 was initiated in March 1990 to correct the wiring deficiency. Work Request (WR) 01389-90 was initiated to implement the design change. The design change was completed on April 12, 1990, and the Halon system returned to an operable status. The licensee also performed a review of schematic diagrams and logic diagrams and found there were no other cases where a similar type auxiliary relay would actuate multiple safety-related trains. These corrective actions appeared to be appropriate to correct the existing condition and determine whether other similar conditions existed. This LER is closed.

3.2 Followup of Previously Identified Items (92702)

(Closed) Violation (482/8905-02): Failure to Lockwire the Turbine Driven Auxiliary Feedwater Pump (TDAFWP) Discharge Isolation Valve in the Neutral Position

On February 8, 1989, an inspector observed that the handwheel to TDAFWP Discharge Isolation Valve AL HV-012 was not lock wired in the neutral position as required by Administrative Procedure ADM 02-102, "Control of Locked Component Status."

The licensee issued Interoffice Memorandum OP 89-0092, on May 24, 1989, which provided a revised locking technique for valves required to be placed in the neutral position. In March 1991, an audit of locked neutral valves was performed by the licensee and no discrepancies were identified. The inspector verified that the handwheels for Auxiliary Feedwater Valves AL HV-6, -8, -10, and -12 were locked in their neutral position in accordance with the revised locking technique.

In May 1991, QA Surveillance TE: 53359 S-1882, "Control of Locked Components," was performed. During the surveillance, the licensee identified 3 of 208 components where the locking device had not been properly installed. The instances identified were the chains on Valves AL-V058 and -V034 (each had a broken chain link), and the chain on Valve AL-V068 could be removed from the handwheel without breaking the green locking tab. Quality Program Deviation 5/91-028 was issued and closed in process because of the prompt corrective action taken to properly install each valve's locking device. On May 6, the licensee examined 20 additional valves and found the locking devices to be properly installed.

On June 6, 1991, with the plant in Mode 1, the inspector observed that the locking chain on TDAFWP discharge to Steam Generator "D" Header Isolation Valve AL-V063 could be removed without breaking the green locking tab. The valve position indicator was positioned at FULL OPEN which indicated that the required flow path was available. Administrative Procedure ADM 02-102, Revision 19, "Control of Locked Components Status," Table 1, page 11 of 47, requires that Valve AL-063 be locked open. The reason provided in the procedure for locking the valve open is "Manual valves in emergency core cooling system flow path that must be OPEN for system operation." The inspector noted that this valve had been included in the May 6, 1991, surveillance. A review of the locked component status log indicated that the valve had been last operated in April 1990. The inspector identified the failure to properly lock Valve AL-V063, such that it could not be repositioned without breaking the green locking tab, as a violation (482/9113-01). This violation is of particular concern because it indicates that previous corrective actions have not been completely effective in correcting similar violations. This violation is also indicative of a weakness with the corrective action verification process because this condition was not detected when it was inspected by licensee personnel in May 1991.

The inspector notified the shift supervisor of the condition and immediate action was taken to properly lock the valve. Out of the total of 12 valves observed, the inspector noted that Valves AL-V049, -V005, and -V058, had used up to four green locking tags each to secure the valve hand wheel. An additional green locking tab was also installed on AL V-063. This practice was also noted during the May audit. Following the May audit, operations personnel issued Letter OP 91-0119 which provides guidance such that, as chains break, the chain locking device should be replaced rather than use an additional green locking tab to restore the locking device's integrity. This use of multiple locking tabs was discussed with licensee management. The licensee is considering replacing the broken chains during the upcoming refueling outage.

(Closed) Violation (482/8839-02): Failure To Take Adequate and Timely Corrective Action for a Diesel Generator Fuel Oil Fill/Vent Line Vibration Failure

This violation was reviewed previously in NRC Inspection Reports 50-482/89-22 and -90-18. The violation remained open pending installation of flexible tubing in the fuel oil fill/vent lines on both diesel generators. This installation activity was completed for the "A" EDG on July 7, 1990, and for the "B" EDG on December 4, 1990. This violation is closed.

(Closed) Violation (482/88200-05): EDG Verification of Seismic and Vibration "As Built" Not Verified

During construction of the EDGs, the licensee had not verified that safety-related seismic and vibration dampening supports on the turbocharger cooling water pipe supports had been installed as required by the vendor's design drawing.

The licensee received the EDG as skid mounted systems from Colt Industries. The subsystems were then assembled in their respective EDG bays. The EDGs were manufactured and furnished under Colt's quality program. This program was reviewed and accepted by WCGS. The licensee's receipt inspection program consisted of: (1) a review of documents submitted; (2) inspection for shipping/handling damage; (3) accountability review for correct material items; and (4) a general configuration review. The responsibility to assure that each component, subcomponent, and sub-subcomponent had been properly installed was left to the vendor's quality program.

The missing pipe supports were subsequently fabricated and installed by the licensee. The licensee has learned that the vendor drawings were provided for the purpose of identifying parts and were not intended to be used as the final "as built" configuration drawings. The licensee then performed a piping system walkdown of the EDGs. Identified discrepancies with the vendor drawings have been properly dispositioned.

Concerns involving licensee/vendor interface were the subject of Violation 482/88200-04. This violation was closed in NRC Inspection Report 50-482/90-04. This violation is closed.

(Closed) Violation (482/9016-01): Change Out of Rod Cluster Control Assembly (RCCA) Tool

On March 29, 1990, the licensee had been performing eddy current testing on an RCCA. The assembly had been placed in its final storage location in the spent fuel pool with the RCCA change tool still connected. Following a shift turnover, an operator proceeded to move the RCCA change tool prior to disengaging the tool from the RCCA. The RCCA change tool was moved approximately 23 inches prior to the operator noting the tool was still attached. No damage occurred to the RCCA.

The licensee's corrective actions included a written reprimand to the operators involved, reemphasizing the importance of proper shift turnovers, and revising Fuel Handling Procedure FHP03-012, "RCCA Change Tool Operating Instructions." The revised procedure requires that an individual observe operation and disengagement of the RCCA change tool. In addition, the revision requires that a senior reactor operator be present during the use of the RCCA change tool. Proper implementation of these revisions appears to be appropriate to prevent recurrence.

During the recent new fuel receipt and unloading conducted in June 1991, in preparation for Refueling Outage V, a manlift contacted a holding clamp on a new fuel assembly. The assembly had been placed in the vertical position in preparation for movement to the new fuel storage pool. The manlift was operated along side the assembly prior to positioning the clamp out of the way of travel. When the manlift contacted the clamp, the clamp bound against a fuel assembly grid strap. No damage occurred to the assembly. (This event is discussed further in paragraph 7 of this report). The inspector discussed the latest event with licensee management to iterate the importance of procedural compliance and assuring that personnel are cognizant of the effects of their actions on plant equipment and other personnel. The licensee's management expressed the same concern and recognized the need to assure that personnel appropriately adhere to procedures during the upcoming refueling outage.

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. During this inspection, the inspectors also reviewed aspects of the fire protection program, medical requirements for licensed operators, compensatory measures for the changeout of the plant computer, and the effects of failed fuel on reactor coolant system (RCS) activity. The methods used to perform this inspection included direct observation of activities and equipment, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and limiting conditions for operations, corrective actions, and review of facility records.

4.1 Fire Detectors Not Calibrated

On May 21, 1991, an I&C technician requested that the surveillance test coordinator extend the annual surveillance interval for calibration of the containment fire detectors. This request was based on an assessment that the detectors were inaccessible during power operations. In reviewing the surveillance test frequencies required by ADM 13-100, Revision 4, "Fire Protection Manual," the surveillance test coordinator discovered that the required surveillance frequency was 6 months. The latest surveillance test schedule required only annual calibration of the detectors. He determined that the surveillance frequencies for fire detector surveillance tests (STN FP-815, -816, -817, -817A through -817F, -818 and -819) had been changed from 6 months to 1 year. These changes were initiated in July 1988 and completed 1 year later. The possibility that the fire detectors had not been calibrated within the last 6 months, as required by the administrative procedure, was brought to the attention of the operations manager at the end of the normal working day. The operations manager then decided to pursue the issue the following day. The decision of management personnel to not immediately pursue the situation to determine the extent of the problem and to make an operability determination is considered a weakness.

At 9 a.m. on May 22, 1991, the supervisor of operations who was on shift as the shift supervisor, the fire protection specialist, I&C personnel, and the manager of compliance held a meeting to discuss the issue that was identified the previous day. I&C was requested to develop a list of detectors that were affected and start performance of the surveillance tests for the detectors that were not calibrated within the past 6 months. As related to the inspector by participants in the meeting, the discussion centered around the requirements of the condition of Facility Operating License No. NPF-42 that requires the determination of the ability to achieve and maintain safe shutdown in the event of a fire. Recent revisions to the National Fire Protection Association Code, which the licensee is committed to, allows annual calibration frequencies for detectors. However, the National Fire Protection Association Code requires detector sensitivity testing prior to implementing an annual calibration frequency. The fire detector sensitivity testing had not been performed by the licensee. Approximately 82 of 149 detectors listed in Table 7.3.4 were not tested within the previous 6 months. Section 7.3.4 of ADM 13-100 requires the fire detection instrumentation for each fire detection zone shown in Table 7.3.4 to be maintained operable whenever equipment protected by the fire detection instrument is required to be operable. Section 7.3.4.3 states that the operability of the fire detection instrumentation shall be demonstrated by testing and surveillance activities. Section 7.3.4.3.1 requires that the operability of accessible fire detection instruments shall be demonstrated at least once per 6 months by the performance of a trip actuating device operational test as detailed in STN FP-815, -816, -817, -817A through -817F, -818, and -819. Failure to calibrate these detectors within 6 months is a violation of TS 6.8.1.h (482/9113-02).

At 3:30 p.m., the licensee issued Fire Impairment 91-169 requiring hourly fire watches for the auxiliary building, control building, radwaste building, and the essential service water intake structure. The control room log noted that the fire detection surveillance frequency required by ADM 13-100 had not been adhered to for an unknown number of fire detectors. Section 7.3.4.2 states that "With more than one-half of the Function A fire detection instruments in any fire zone shown in Table 7.3.4 inoperable or with any Function B fire detection instruments shown in Table 7.3.4 inoperable, or with any two or more adjacent fire detection instruments inoperable, within 1 hour establish a fire watch patrol to inspect the zones with the inoperable fire detection instruments at least once per hour, unless the instrument is located inside the containment, then inspect that containment zone at least once per 8 hours or monitor the containment air temperature at least once per hour at the locations listed in WCGS's TS 4.6.1.5."

The requirement in Section 7.3.4.2 to monitor containment temperature was initially misinterpreted as being required within 8 hours. The containment temperature monitoring was not performed hourly, within an hour of identification, in accordance with the procedure until 6:30 p.m. This is the second example of Violation 482/9113-02.

On May 23 at 11:20 a.m., the licensee notified the NRC Operations Center, within the 24-hour time requirement, of a potentially reportable violation of License Condition 2.C.(5)(b) and initiated a programmatic deficiency report (PDR). The license condition states that "The licensee may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire." This condition was incorporated into the TS on February 24, 1988.

The license condition was issued to allow the licensee to make changes based on the performance of an evaluation to determine if changes would adversely affect the ability to achieve and maintain safe shutdown in the event of a fire. The lack of administrative controls to prevent changes to the fire protection program without performing an evaluation is considered a weakness. The lack of administrative controls to control surveillance test frequencies to ensure compliance with administrative procedures is also considered a weakness.

I&C personnel, working two 12-hour shifts, completed the performance of the fire detector calibrations on May 24, 1991. Because no detectors were found out of calibration, the licensee determined that they were not in violation of their license condition since there would have been no adverse effect on the ability to achieve and maintain safe shutdown in the event of a fire. Therefore, the licensee determined that no followup report was required to be submitted.

On May 29, 1991, ADM 13-100 was revised to denote the four smoke detectors in containment as being inaccessible during power operation. These detectors are noted as Function A (early warning fire detection and notification). Discussions with licensee personnel revealed that there are currently no plans to increase the fire detector surveillance frequency from 6 months to annually.

On June 26 a draft of the PDR was provided to the inspector. The PDR was still under evaluation and had not been entered into the formal PDR tracking system. The inspector found the delay in entering this PDR into the formal PDR tracking system to be excessive. This delay did not ensure that management was cognizant of the PDR status.

4.2 Inoperable Fire Suppression System

On May 22, 1991, the inspector noted that the pressure of the fire suppression Halon bottle for the control room cable chases was below the required pressure to ensure adequate suppression capability. Surveillance STN-FP-404, Revision 1, "Halon System Tank Weight and Pressure," is a 6-month surveillance that was last performed on January 18, 1991. During the surveillance, the Halon bottle was replaced because of low system pressure. The control room cable chase Halon bottles were not included on operator logs. Other Halon systems are included in the operator log sheets requiring verification of adequate system pressure on a daily basis. Section 7.3.5 of ADM 13-100, Revision 4, "Fire Protection Program Manual," requires, in part, that the Halon system for the control room cable chases shall be maintained operable whenever equipment protected by the Halon system is required to be operable. The failure to maintain the control room cable chase Halon bottle operable is the third example of Violation 482/9113-02.

The inspector notified a fire protection staff member who immediately initiated a WR to change out the Halon bottle. A fire impairment was also initiated to designate control room personnel as continuous fire watches in accordance with ADM 13-100. The Halon bottle was changed out on June 5, 1991.

4.3 Discrepancies in Medical Examinations

On May 21, 1991, the licensee notified the inspector that a recent QA audit of licensed operator medical examinations had identified some discrepancies in the current program that implements the requirements of American National Standards Institute (ANSI) American Nuclear Society (ANS) 3.4-1983 and 10 CFR Part 55.53(a)(1). The audit was performed at the request of the training manager to ensure that problems noted at other facilities were not present in the WCNOC program. The discrepancies include the failure to perform a specific test to determine tactile discrimination capability, blood tests to document the absence of hematopoietic dysfunction, and procedural requirements to review the status of work performance, attendance, and behavioral changes that are documented as part of the fitness-for-duty program. Although these requirements were not met, the licensee stated that they had reasonable assurance that all licensed operators were medically qualified on the basis of other examinations that met the intent of ANSI/ANS 3.4-1983. The licensee committed to strengthen their program in these areas and perform the required testing. This will remain an unresolved item pending the results of the required testing (482/9113-03).

4.4 Trip of EDG Output Breaker

On June 14, 1991, during the transfer of the 4160-volt emergency Bus NB01 supply from SL-7 to No. 7 transformer, the "A" EDG output breaker was closed and the power-factor meter went off scale high. When the operator tried to restore the power factor to 0.9, the EDG reversed power and the output breaker tripped. The licensee was reviewing this event to determine whether this constituted an invalid failure of the "A" EDG. When a second attempt to load the EDG was made, the power-factor meter stayed in the normal range, and the EDG was successfully loaded.

4.5 Leak-Rate Test for Containment Auxiliary Access Hatch Door

On June 18, 1991, during a maintenance work scheduling meeting, the licensee discussed the need to perform a leak-rate test, whenever the auxiliary access hatch is open. This hatch consists of two doors through which emergency egress can be made from containment and is part of the containment boundary.
TS 4.6.1.3 requires that a leak-rate test be performed within 72 hours after

each time the hatch door is open. Following the meeting, electrical maintenance personnel contacted results engineering and informed them that a leak-rate test was not performed following an entry into the auxiliary access hatch on May 9, 1991. Electrical maintenance personnel had entered the hatch from outside containment to work the alarm system for the security door that allows access to the auxiliary access hatch door from outside containment. This work was performed under a security system WR. These WRs are not normally reviewed at the plan-of-the-day meeting. Entry into this area requires that health physics personnel check out a high radiation area key from security. Security department personnel are then required to notify the shift supervisor. The checkout of a high radiation area key is normally included in the shift supervisor's log; however, the log entry did not discuss the reason for the entry into the auxiliary access hatch. The opening of the auxiliary access hatch door does not appear to have been discussed with the shift supervisor prior to or following entry. The licensee's immediate corrective action was to discuss the requirement with all electrical maintenance personnel. An LER will be issued to address the root cause of the event and any additional corrective actions. The inspector noted that additional work performed on the hatch door on June 26, 1991, was addressed in the shift supervisor's log and results engineering was notified to perform the required leak-rate test. This event will be reviewed further following the issuance of the LER.

4.6 Nuclear Plant Information System (NPIS) Computer

On June 18, 1991, OFN 00-023, Revision 7, "Loss of NSSS/BOP Computer," was superseded by OFN 00-023, Revision 8, "Loss of NPIS Computer." The increased awareness tours were for the turbine building and auxiliary building, and site watches were discontinued.

4.7 Effects of Failed Fuel on RCS Activity

Increased levels of dose equivalent iodine and RCS gross activity continued during the inspection period. Action Level 2 of the Failed Fuel Action Plan was entered May 25, 1991. As a result, letdown was increased to 120 gallons per minute which required the use of the centrifugal charging pump instead of the positive displacement pump. At the end of the inspection period, the licensee was planning to operate the boron thermal regeneration system to remove boron from the RCS to maintain the current power level. Other planned actions included degassing of the volume control tank to reduce RCS activity levels.

Conclusions

There were three examples of failure to properly implement the fire protection program. An unresolved item pertaining to the resolution of discrepancies associated with licensed operator medical examination requirements was identified. There were apparent weaknesses to make a prompt determination of operability of fire detectors that were outside of the required calibration frequency and to promptly document a condition adverse to quality (i.e., failure to properly implement compensatory actions when fire detectors were determined to be inoperable). Action Level 2 of the failed fuel action plan was entered.

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. Methods used to perform this inspection included direct observation of licensee activities and review of records.

5.1 STN IC-218A, Revision 0, "DLG Room 'A' Temperature Sensor TE-1"

On June 26, 1991, STN IC-218A, Revision 0, "DG Room 'A' Temperature Sensor TE-1 and Ventilation Controls," was performed to adjust the setpoint and the proportional gain for the "A" EDG room recirculation and outside air dampers. The "A" EDG was declared inoperable at the start of the surveillance at 3:30 p.m. During the performance of the surveillance, an error was noted in the tolerance range of the voltage to be read. The specified voltage was 0.0 with a tolerance band of -0.5 to +0.5. The tolerance range should have been from -0.05 to +0.05. The error appeared in two places in the procedure. The surveillance was delayed while a procedure change was written. The procedure for the "B" EDG, STN IC-218B, Revision 1, which was to be performed subsequent to the "A" test, did not contain the same error. Setpoint Change Request GM 91-068, performed with this procedure, required the adjustment of the proportional band. Insufficient information was included with the document to indicate where in the cabinet the potentiometer was located. In order to locate the potentiometer, a call had to be placed to personnel who were more familiar with the equipment. As a result of procedural weaknesses, the "A" EDG was inoperable approximately 1 1/2 hours longer than for the performance of the same procedure for the "B" EDG. The inspector reviewed the last four performances of STN IC-218A. Two tests required only partial completion and did not utilize the steps that contained the erroneous tolerance values. The surveillances performed on May 3, 1991, and February 7, 1991, were found to have "as found" and "as left" tolerances within the correct tolerance band of 0.05 volts. It was not evident in looking at the procedures if the performer was cognizant of the error. Revision 0 of this procedure was approved in August 1985. Subsequent 2-year reviews have also failed to detect the error in tolerance range.

5.2 EDG Room Temperature Profile

On June 4, 1991, the licensee performed a temporary procedure (TP) to determine the actual temperature profile in an EDG bay with an EDG running and the ventilation system shut down. The test was performed in accordance with TP TS-38, Revision 0, "EDG Room Temperature Profile." This TP was initiated to determine if the ventilation system was required to be operable while the EDG was running. LER 91-04 documents several occasions where the EDGs were declared operable with the ventilation system inoperable, a violation of the TS.

Prior to the test, the precautions and limitations were discussed between the operators and results engineering. The licensee established that the ventilation system would be started when EDG bay temperatures approached 115°F and that the EDG would be declared inoperable prior to bay temperatures reaching the TS limit of 119°F. Initial conditions within the EDG "A" bay were

86.4°F and 43.5 percent relative humidity. The EDG "A" was then started with the ventilation fan in pull-to-lock. After approximately 4 hours, temperatures within the bay continued to increase, indicating that the TS temperature limit would be reached with continued operation. The ventilation fan was then started and the bay temperatures were decreased to well below the TS limit.

A previous test performed with the initial temperature at approximately 67°F resulted in the bay coming to an equilibrium temperature below 115°F. On the basis of the test results, the licensee has determined that the ventilation system is required to be operable for EDG operability.

5.3 Additional Surveillance Testing

The following completed surveillance test records were also reviewed and/or witnessed:

- STS KE-003, Revision 8, "Spent Fuel Pool Cranes Surveillance Test;"
- STS SE-002, Revision 4 and MI 91-444, "Manual Calculation of Reactor Thermal Power;"
- STS BB-004, Revision 7 and MI 91-441, "RCS Water Inventory Balance;"
- STN KJ-001, Revision 3, "D/G Rocker Arm Prelube Oil Pump Operation;"
- STS NB-005, Revision 6, "Breaker Alignment Verification;"
- STS AB-201, Revision 10, "Main Steam System Inservice Valve Test for AB HV-005 only;" and
- STS PE-013, Revision 9, "Personnel Air Lock Seal Test."

Conclusions

Surveillance tests observed were performed in accordance with the approved procedures. However, inaccuracies and lack of detail in one procedure resulted in the "A" EDG being out of service for approximately 1 1/2 hours longer than the "B" EDG.

6. MONTHLY MAINTENANCE OBSERVATIONS (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 observation, personnel interviews, and records review. Portions of selected maintenance activities regarding the WRs were observed. The following WRs and related documents are reviewed by the inspectors.

6.1 Replacement of Main Feedwater Pump Governor Control Power Supply

On June 17 the 15-volt power supply associated with the control circuit for the governor of the "A" main feedwater pump failed; however, the backup power supply assured continued operation of the pump. WR 02384-9' was written to implement Temporary Modification 91-31-FC that installed a 15-volt power supply exterior to the pump control cabinet. Some problems were experienced with the comparator circuit board. The final installation was viewed by the inspector. A fan was placed near the power supply to ensure cooling of the power supply.

6.2 Inadequate Clearance Order Implementation

During a review of active clearance orders, operations personnel discovered that Clearance Orders 91-449-WT and 91-824-WL had been accepted without the 480-volt breakers documented as being deenergized. Electrical maintenance was contacted, the breakers were verified as deenergized, and a PDR was written. Although these clearance orders did not affect safety-related equipment, the potential for impacting personnel safety did exist.

6.3 Additional Maintenance Testing

The inspector observed the performance of the following maintenance activities and verified that the WRs were properly documented.

- WR 51191-01 Cleaning and inspection of load center transformer for B/U pressurizer heaters (XPG-21);
- WR 51162-91 PG-21 main supply breaker;
- WR 51161-91 Load center 480 VAC;
- WR02160-91 Spent fuel pool pump feeder breaker-replace nephrine buffer assembly; and
- WR 51292-91 SFP pump breaker PMs.

Conclusions

Maintenance and modifications were performed well. A concern was noted relating to the inadequate adherence to procedures in establishing a clearance order.

7. PREPARATION FOR REFUELING (60705) (93702)

New fuel was delivered, offloaded, inspected, and placed in the spent fuel pool during the inspection period. The inspectors observed portions of several fuel receipt inspections performed in accordance with FHP 01-001, Revision 13, "New Fuel Receipt." On June 5, 1991, after the removal of the first fuel assembly from the shipping cask, the manlift was raised prior to the clamping frames for the removed assembly being closed as required by Step 7.2.10 of FHP 01-001. The manlift contacted one of the extended clamping frames, resulting in a force on the clamping frame of the fuel assembly that was still in the support frame. This resulted in contact with the second fuel assembly grid strap. Later inspection revealed no damage to the grid strap. The corrective action taken was to stop fuel movement and modify the procedure to include the requirement to perform a briefing with all personnel involved in new fuel receipt prior to fuel movement.

Earlier the same day, two other PDRs were generated. The licensee initiated one PDR for failing to follow procedures when the tape around the bottom of the fuel assembly was broken when it should have been left in tact because of inconsistencies between QC Procedure 7.1 and FHP 01-001. The second PDR was initiated because the crane operator failed to follow Procedure Step 7.1.15, which required placing the normal/jog control switch in jog position prior to loading 100-200 pounds with the monorail hook.

These PDRs had not been entered into the tracking system at the end of the report period.

Conclusions

Problems occurred which could have resulted in the damage to new fuel. Similar problems were noted with procedure compliance during new fuel receipt prior to Refuel IV. Procedure FHP 01-001 was revised to include the requirement for a briefing of personnel involved with fuel handling by reactor engineering prior to fuel movement. A weakness was noted in failing to assign tracking numbers to PDRs.

7. EXIT MEETING

The resident inspector met with licensee personnel (denoted in paragraph 1) on July 1, 1991. The inspector 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

ANS	American Nuclear Society
ANSI	American National Standards Institute
CCW	component cooling water
DG	diesel generator
EDG	emergency diesel generator
ESF	engineered safety feature
I&C	instrumentation and control
LER	licensee event report
NPIS	Nuclear Plant Information System
NRC	Nuclear Regulatory Commission
NPE	nuclear plant engineering
PDR	programmatic deficiency report
QA	quality assurance
RCCA	rod cluster control assembly
RCS	reactor coolant system
STN	surveillance nontechnical specification
STS	surveillance technical specification
TDAFWP	turbine driven auxiliary feedwater pump
TP	temporary procedure
TS	Technical Specification
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation
WR	work request