APPENDIX B

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

NRC Inspection Report: 50-482/88-07 Operating License: 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: February 15 through March 31, 1988

Inspectors: Dwight D. Chamberlain for 5-5-88
B. L. Bartlett, Senior Resident Inspector, Project Section A, Division of Reactor Projects — Date

Dwight D. Chamberlain for 5-5-88
M. E. Skow, Resident Inspector, Project Section A, Division of Reactor Projects — Date

Approved: Dwight D. Chamberlain 5-5-88
D. D. Chamberlain, Chief, Project Section A, Division of Reactor Projects — Date

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Inspection Summary

Inspection Conducted February 15 through March 31, 1988 (Report 50-482/88-07)

Areas Inspected: Routine, unannounced inspection including followup of previously identified NRC items, review of "Safety Outage Modifications Inspection" items, operational safety verification, engineered safety features system walkdown, monthly maintenance observation, monthly surveillance observation, onsite event followup, radiological protection, and physical security.

Results: Within the nine areas inspected, one violation (failure to follow procedures for modification and maintenance activities, paragraphs 4 and 6) was identified.

DETAILS

1. Persons Contacted

Principal Licensee Personnel

*B. D. Withers, President and CEO
*R. M. Grant, Vice President, Quality Assurance (QA)
*J. A. Bailey, Vice President, Engineering & Technical Services
*G. D. Boyer, Plant Manager
*O. L. Maynard, Manager, Licensing
C. M. Estes, Manager, Operations
R. Hollaway, Manager, Maintenance
M. G. Williams, Manager, Plant Support
C. E. Parry, QA Manager
*A. A. Freitag, Manager, Nuclear Plant Engineering (NPE)
K. Peterson, Licensing
*G. Pendergrass, Licensing
*W. M. Lindsay, Manager, Quality Evaluations
*C. J. Hoch, QA Technologist
J. Goode, Licensing Engineer
B. McKinney, Manager, Technical Support
*R. Flannigan, Supervisor, Compliance

The NRC 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 March 30, 1988.

2. Followup on Previously Identified Inspection Findings (92701)

a. (Closed) Open Item (482/8634-04): Inservice Inspection Of Safety Valves - This item concerned the retest schedule of the three pressurizer code safety valves. Discussions with NRC personnel knowledgeable of the code testing requirements has resulted in the conclusion that the licensee meet the code testing requirements. All three valves were retested during the end of cycle two refueling outage. This item is closed.

b. (Closed) Open Item (482/8727-07): Inspection of Piping - The licensee committed to NRC that prior to returning to operation following the 1987 refueling outage, ultrasonic (UT) inspection of all safety-related piping locations having a high-probability of erosion/corrosion damage, would be completed. The NRC inspector, through interviews of licensee personnel, observation of UT inspections, and review of documentation, verified the performance of the inspections. This item is closed.

c. (Closed) Open Item (482/8727-08): Replace Eroded/Corroded Piping - The licensee committed to NRC that prior to returning to operation following the 1987 refueling outage, areas of piping which has experienced through wall or below minimum wall thinning, would be replaced with stainless steel components. The NRC inspector verified that the licensee replaced four areas eroded below minimum wall. A fifth area was replaced even though it was not required. Two areas that were originally going to be replaced were not required to be. This item is closed.

d. (Closed) Open Item (482/8727-09): Replace Piping Projected To Erode/Corrode Below Minimum Wall - The licensee committed to NRC that prior to returning to operation following the 1987 refueling outage, all components projected to erode/corrode to less than minimum wall thickness during the third fuel cycle will be repaired or replaced. The licensee did not find any safety-related piping that would be below minimum wall during this fuel cycle. Two nonsafety-related areas were found and corrected. This item is closed.

e. (Closed) Open Item (482/8727-10): Provide a Plan and Schedule - The licensee committed to provide for NRC review by November 30, 1987, a plan and schedule for corrective measures that will eliminate erosion/corrosion related pipe thinning caused by flow disturbances in piping systems as a result of butterfly valve throttling and piping configurations. Licensee Letter ET 87-0362 dated November 30, 1987, supplied the requested information to NRC. This item is closed.

f. (Closed) Open Item (482/8806-02): Component Cooling Water Water-Hammer Event - This item was open pending further review of the licensee's corrective and followup actions to verify system operability subsequent to the water-hammer event. The licensee's actions were reviewed and reported in paragraph 3 of NRC Inspection Report 50-482/88-08. In that report, the licensee's actions appeared to be acceptable. Based on that report and upon discussions with the NRC inspectors involved, this item is considered closed.

3. Review of "Safety Outage Modifications Inspection" Items

On February 8, 1988, NRC Inspection Report 50-482/87-32, "Safety Outage Modifications Inspection," was issued. The NRC senior resident inspector (SRI) reviewed this report and unresolved item numbers are being generated as follows to ensure proper close out of all safety outage modifications inspection (SSOMI) findings.

a. (Open) Unresolved Item (482/8807-01) (paragraph 2.1.2.1): PMR 2024 Battery Charger AC Alarm Setpoint - The impedance data used as a calculational input for one of the safety-related load center transformers was based upon General Electric test data; however, this impedance value did not agree with the transformer's nameplate data.

b. (Open) Unresolved Item (482/8807-02) (paragraph 2.1.2.2.a): PMR 899 Accumulator Level Transmitters - As a result of this PMR, a smaller volume of nitrogen gas remained in the accumulator tank to provide for water injection into the primary system in the event of a loss of coolant accident.

c. (Open) Unresolved Item (482/8807-03) (paragraph 2.1.2.2.b): PMR 899 Accumulator Level Transmitters - Although a significant amount of data existed to justify this PMR, it was not communicated within the organization. A root cause analysis for changing the level transmitters was not performed.

d. (Open) Unresolved Item (482/8807-04) (paragraph 2.1.2.2.b): PMR 899 Accumulator Level Transmitters - PMR 899 failed to have the Q-list of all safety-related equipment revised to reflect the changes brought about by the modification.

e. (Open) Unresolved Item (482/8807-05) (paragraph 2.1.2.3): PMR 2167 Electrical Equipment Room No. 1403 Chiller - A field change request (FCR) which was issued failed to fully evaluate the effect of the FCR. The installation of a manual valve was inadequate because a temperature indicator was not provided to measure the temperature of Room 1403, the TS surveillance procedures did not include this room for periodic surveillance and the room was not required to be monitored for environmental conditions. In addition, the basis for the acceptability of 75°F was not addressed.

f. (Open) Unresolved Item (482/8807-06) (paragraph 2.1.2.4.a): PMR 1634 Reactor Coolant Drain Tank (RCDT) Isolation Valve - The safety evaluation performed for this modification failed to show that with one relief valve gagged, that the remaining relief valve provided equivalent or adequate protection for the tank.

g. (Open) Unresolved Item (482/8807-07) (paragraph 2.1.2.4.b): PMR 1634 RCDT Isolation Valve - The new isolation valve added by PMR 1634 had less flow area than the relief valve inlet, contrary to the requirements of paragraph UG-135, Appendix M, Section VIII, of the ASME Boiler and Pressure Vessel Code Division 1-1974. An analysis to verify that the isolation valve would not reduce the capacity of the relief valve was not performed.

h. (Open) Unresolved Item (482/8807-08) (paragraph 2.1.2.4.c): PMR 1634 RCDT Isolation Valve - Instrumentation was not installed at the new isolation valve to enable appropriate emergency actions if the tank became overpressurized.

i. (Open) Unresolved Item (482/8807-09) (paragraph 2.1.2.5): PMR 1613 Valve Leakoff Configurations - This PMR added flexible hoses and a shutoff valve on the leakoff lines to certain valves. The manufacturer's rated pressure for the hoses is less than the pressure that the hose could be subjected to.

j. (Open) Unresolved Item (482/8807-10) (paragraph 2.1.2.7.a): PMR 2206 Auxiliary Building Fire Detection System - This PMR made modifications and authorized an increase in the fire loading below the auxiliary feedwater pump (AFWP) rooms. However, no fixed water suppression systems were installed.

k. (Open) Unresolved Item (482/8807-11) (paragraph 2.1.2.7.b): PMR 2206 Auxiliary Building Fire Detection System - Uncovered cable trays containing the power cables for both the motor driven AFWP's pass vertically through the area addressed by this PMR with less than 20 feet separation from combustible material.

l. (Open) Unresolved Item (482/8807-12) (paragraph 2.1.2.7.c): PMR 2206 Auxiliary Building Fire Detection System - The installation of two storage areas in close proximity to safety-related systems appears to violate guidelines in the Updated Safety Analysis Report (USAR).

m. (Open) Unresolved Item (482/8807-13) (paragraph 2.1.2.8): PMR 2222 Containment Cooling Fan Damage - This PMR implemented corrective actions following the failure of a fan blade in Containment Fan Cooler SGN01B. The licensee failed to investigate and determine the root cause of the failure.

n. (Open) Unresolved Item (482/8807-14) (paragraph 2.1.2.9): Appendix J Leak Test Requirements - Further NRC action is necessary to clarify the containment boundary and leak testing requirements with respect to the secondary systems, the assumed condition of the secondary systems during a design basis accident, and the design of the secondary systems with respect to high energy line breaks. This concern has been identified at other plants.

o. (Open) Unresolved Item (482/8807-15) (paragraph 2.1.2.10): Battery Discharge of October 15, 1987 - The licensee indicated that operational procedures were not routinely utilized for removing and returning equipment to service. The licensee should have provided adequate controls for the removal from service of a safety system which resulted in the loss of the station batteries.

p. (Open) Unresolved Item (482/8807-16) (paragraph 2.1.2.10): Battery Discharge of October 15, 1987 - The operations department failed to use battery sizing calculations provided by engineering and failed to consult engineering. This failure to involve engineering is a significant weakness.

q. (Open) Unresolved Item (482/8807-17) (paragraph 2.1.2.11): Battery Sizing Calculation E-3 - The review of Bechtel Battery Sizing Calculation E-3, "Class 1E Battery System," Revision 0, identified discrepancies.

r. (Open) Unresolved Item (482/8807-18) (paragraph 2.1.2.12): Battery Performance Test - The review of Surveillance Test STS MT-022, performed on Battery NK12 on October 7, 1987, identified weaknesses. There was a failure to recognize that the battery would recover lost capacity during extended delays in testing.

s. (Open) Unresolved Item (482/8807-19) (paragraph 2.1.2.13): DC System Low Voltage Alarms - The four DC undervoltage alarms which existed in each battery system should have provided sufficient warning to have prevented the low voltages experienced on October 15, 1987.

t. (Open) Unresolved Item (482/8807-20) (paragraph 2.1.2.14): Diesel Generator Breaker Operations - The SSOMI team identified a potential problem with the design of the closing circuit for the emergency diesel generator output breakers. Under certain circumstances the "anti-pumping" logic could prevent the breakers from closing onto a cleared, deenergized bus without operator action.

u. (Open) Unresolved Item (482/8807-21) (paragraph 3.1.2.2.a): PMR 1722 Motor-Operator Testing - The spare conductor in Valve BB-HV8000A was unprotected and the spare conductor in Valve BB-HV8000B and its tape had become unraveled. Drawing E-11013 allows the use of tape for the protection of conductors outside, but not inside, the reactor building.

v. (Open) Unresolved Item (482/8807-22) (paragraph 3.1.2.2.6): PMR 1722 Motor-Operator Testing - Engineering Evaluation Request (EER) 86-EM-03 described the problem of multiple drawings and inadequate cross referencing between plant and vendor drawings. This EER had not been evaluated or dispositioned.

w. (Open) Unresolved Item (482/8807-23) (paragraph 3.1.2.2.b): PMR 1722 Motor-Operator Testing - EER 87-KC-08 was written to correct a vendor wiring drawing that used the same wire number twice. The resolution to this EER was considered inadequate.

x. (Open) Unresolved Item (482/8807-24) (paragraph 3.1.2.4.a): PMR 2018 ASCO Solenoid Valve Replacement - The 10 CFR 50.59 safety evaluations did not document addressing seismic and environmental equivalency of the new valves.

y. (Open) Unresolved Item (482/8807-25) (paragraph 3.1.2.4.a): PMR 2018 ASCO Solenoid Valve Replacement - The air supply line for ASCO Solenoid Valve EJ HCV-8890B appeared to have inadequate seismic support between the solenoid valve and the polar crane wall. In addition, the 3/8-inch tubing supplying air to the valve had one loose support.

z. (Open) Unresolved Item (482/8807-26) (paragraph 3.1.2.5.a): PMR 2329 Raychem Splices - The PMR did not contain an evaluation or documentation which indicated that the WCGS design LOCA environment is equivalent to or less severe than the design LOCA environment that was used as a basis to accept the use-as-is disposition.

aa. (Open) Unresolved Item (482/8807-27) (paragraph 3.1.2.5.b): PMR 2329 Raychem Splices - Documentation that each splice listed on Work Request 4443-87 has a bend radius greater than the bend radius tested by the laboratory was not available. Confirmation that each of the splices conformed with the results of the laboratory test should be documented.

bb. (Open) Unresolved Item (482/8807-28) (paragraph 3.1.2.5.c): PMR 2329 Raychem Splices - The Raychem splices performed on this PMR were bent after the tubing had cooled to room temperature after being heat shrunk. The test reports documented that the tubing was bent while it was heated. Bending the cooled Raychem tubing is possibly less conservative because of the reduced pliability of the tubing at lower temperature.

cc. (Open) Unresolved Item (482/8807-29) (paragraph 3.1.2.6): PMR 1828 ESW Building Cable Replacement - This PMR was issued to pull new cables to the ESW Building. The licensee performed an evaluation of the damaged cables; however, an additional evaluation to determine the root cause of the failure in the original safety-related cables routed to the ESW building was not performed. The SSOMI was concerned that the conditions associated with the original cable pulls were not evaluated to provide assurance that the cable failures were not the result of a generic condition.

dd. (Open) Unresolved Item (482/8807-30) (paragraph 3.1.2.9): Technical Specification Tests - The SSOMI team identified several examples of what they considered to be weaknesses in Surveillance Tests STS IC-433. The SSOMI team was concerned that detailed and accurate test instructions were not provided to ensure that test objectives and applicable TS requirements are fulfilled in approved surveillance procedures.

ee. (Open) Unresolved Item (482/8807-31) (paragraph 3.2.2.1): Temporary Modification TMO 87-120 GK - This modification clamped the Train A Control Room Emergency Ventilation System supply damper in the open, actuated position in response to several failures of the actuating linkage. The SSOMI team determined that the licensee had not adequately determined whether the system remained operable and capable of pressurizing the control room as required by TS 3/4.7.6. The SSOMI team concerns with this modification are listed below:

- The licensee had not performed a calculation or a functional test to demonstrate the ability of the Train B CRVIS to maintain the required control room pressure in this degraded mode.

- The need for operator action to meet the single failure design of a safety system does not conform to the requirements of 10 CFR 50, Appendix A.

- Even though the operator actions did not meet single failure design requirements, the specified operator actions would not be sufficiently responsive when considering the design requirement of the CRVIS to maintain a positive pressure in the control room in the event of radiation or gas in the air intakes.

- The licensee failed to recognize that additional testing or calculations were necessary to verify that the system remained operable with the temporary modification implemented.

- Appropriate corrective action in preventing repeated damper failures was not taken.

- The licensee had not identified the above failure as potentially reportable nor evaluated the failures for reportability.

ff. (Open) Unresolved Item (482/8807-32) (paragraph 3.2.2.2): PMR 2106 Pressurizer Spray Valve Bonnet Repair - The SSOMI teams review of EER 85-XX-37 identified the concerns listed below:

- Section 3.2, "Bolted Connection," of the application procedure, which was detailed in the engineering disposition to EER 85-XX-37 and used to repair the Pressurizer Spray Valve, was not verified to meet the ASME Code requirements.

- The SSOMI team considered that the 10 CFR 50.59 Safety Evaluation of EER 85-XX-37 was inadequate, in that it did not identify that drilling holes into pressure retaining parts of ASME Class 1 components involved changes to the facility.

- The maximum pressure used to inject the sealant compound in the body-to-bonnet area of the pressurizer spray valve was not recorded.

- A safety evaluation was not performed for the vendor work procedure to repair the pressurizer spray valve by sealant injection as required. ADM 07-100 requires that a 10 CFR 50.59 Safety Evaluation be completed for all procedures reviewed by the plant safety review committee (PSRC).

gg. (Open) Unresolved Item (482/8807-33) (paragraph 3.2.2.3): PMR 2084 CCW Pipe Wall Thinning - This PMR involved application of the weld overlay on a component cooling water (CCW) line servicing the Train A Residual Heal Removal (RHR) Heat Exchanger in order to repair a pipe section downstream of Valve EJ-V033. The SSOMI team's review of this PMR identified the concerns listed below:

- The licensee did not declare Train A of the CCW and RHR systems inoperable per TS after identifying that the piping was below minimum wall thickness.

- The procedures used permitted welding to occur under conditions that were considered to be nonconservative.

- Major changes in the work instructions for weld overlay repairs on the EJ-V033 piping were implemented by Revision 2 to WR 0702-87, but were not incorporated into the revised ASME Section XI work instructions as required by Section 9.3.6.1 of ADM 01-036, "WCGS ASME Section XI Repair and Replacement Program," Revision 2.

- The repairs were completed without craft or QC signoff of the revised work instructions.

- The post-modification leak testing of the piping was not accomplished as required by procedure until 2 months following completion of the work.

- During the review of this PMR, the SSOMI team also noted that ASME required radiography was incorrectly deferred by the engineering disposition of Corrective Work Request (CWR) 0702-87, Revision 1, for about 6 months.

hh. (Open) Unresolved Item (482/8807-34) (paragraph 3.2.2.4): PMR 2116 Hard Surfacing of Certain Valves - The SSOMI team's review of testing requirements identified that the hydrostatic test instructions for replacement of ESW piping downstream of Valve EF-V58, did not consider possible over-pressurization of adjacent systems.

ii. (Open) Unresolved Item (482/8807-35) (paragraph 3.2.2.6): PMR 0904 Essential Service Water Check Valves - The review of the PMR and the associated work packages found that Clearance Order No. 871061EF, which was used to establish the clearance boundaries for the work, did not provide correlation between the initials and signatures of the individuals establishing and verifying the boundary as required by 10 CFR 50, Appendix B, Criterion XVII, "Records."

jj. (Open) Unresolved Item (482/8807-36) (paragraph 3.2.2.7): PMR 1363 Charging Pump Control Valve Cavitation Damage - WR 4430-86, used to implement PMR 1363 for Valve FCV-121 did not note that PMRs 1613 and 1635 were performed concurrently. The failure to annotate WR 4430-86 to cross-reference the above PMRs for documentation completeness is considered a weakness.

kk. (Open) Unresolved Item (482/8807-37) (paragraph 3.2.2.9): Pressurizer Safety Valve Testing - The SSOMI team reviewed the ASME Code Section XI testing of the pressurizer safety valves. The test procedure and the implementation of the test procedure were inadequate because the test procedure did not use the representative temperature of the valve when installed in the system as required by TS 3/4.4.2.2. Additionally, because the "as-left" valve temperature was not recorded for valves which were tested and reset, the TS requirements could not be substantiated. The concerns listed below were also identified:

- Five of the six valves tested had setpoints well below the minimum TS limit of 2461 psig.

- The safety valves exhibited greater than expected setpoint deviation with respect to temperature variations.

- The temperature conditions used to test the installed valves were different from the temperature conditions used to test the spare valves. Additionally, the test temperature data was not required nor collected for the valves that were tested and reset and the licensee could not substantiate that the test temperatures represented the "as-installed" valves ambient conditions as required by TS 3/4.4.2.2.

- NPE had not been involved in the evaluation of data nor the determination of the correct test temperatures. As a result of the SSOMI team concerns, the maintenance department issued EER No. 87-BB-21 on November 16, 1987, requesting specification of a valve test temperature range which would satisfy the TS requirements. The disposition to this EER, issued on November 18, 1987, provided a temperature range of 70 to 120°F and direction for obtaining temperature measurements. The SSOMI team considered that the EER disposition was unsatisfactory because the temperature recommended by NPE did not have a correlation with the actual installed conditions.

- The SSOMI team found that the ring settings for the spare pressurizer safety valves were not set as required by the manufacturer to ensure a proper valve blowdown characteristic. Furthermore, the guide ring position had not been verified on the installed pressurizer safety valves, and the testing procedures did not include steps to verify guide ring position whenever the pressurizer safety valves were tested.

- The licensee failed to evaluate the information contained in I&E Information Notice 86-05, "Main Steam Safety Valve Test Failures and Ring Setting Adjustments," and I&E Information Notice 86-05, Supplement 1, which identified valve performance problems resulting from improperly established guide ring settings.

4. <u>Operational Safety Verification (71707)</u>

The NRC inspectors verified that the facility is being operated safely and in conformance with regulatory requirements by direct observation of licensee facilities, tours of the facility, interviews and discussions with licensee personnel, independent verification of safety system status and limiting conditions for operations, and reviewing facility records.

The NRC inspectors, by observation of randomly selected activities and interviews of personnel verified that physical security, radiation protection, and fire protection activities were controlled.

By observing accessible components for correct valve position and electrical breaker position, and by observing control room indication, the NRC inspectors confirmed the operability of selected portions of safety-related systems. The NRC inspectors also visually inspected safety components for leakage, physical damage, and other impairments that could prevent them from performing their designed functions.

Selected NRC inspector observations in this area are discussed below:

- A fire impairment control permit was noted in the turbine driven auxiliary feedwater pump room. Permit No. 88-69 appeared to require extra extinguisher/firehose in the area because an adjacent block was checked on the form. The NRC inspector did not observe any extra extinguishers or firehoses in the area. This was discussed with the shift supervisor. He stated that although the block was checked, no extra fire equipment was intended. He explained that he thought that since the block was above a line dealing with TS and that line was applicable in this case, the block should be checked. The NRC inspector reviewed the other impairment control permits in effect and noted one other that referenced the same TS and had the block checked. It was also noted that at least three other permits noted the same TS and did not have the block checked. Procedure ADM 13-103 was reviewed and was found silent in this aspect of completing the permit form. This was discussed with the fire protection engineer. He stated that he was already working on a change to ADM 13-103 and that this would be clarified then. He also stated that he would provide guidance in the interim for the shift supervisors. A memo from the fire protection engineer dated February 23, 1988, was issued, and a copy placed with the impairment control permit log.

- Scaffold storage was observed in the auxiliary building at two locations. One area was at the 1974-foot level north end. The other storage area was the Control Room Ventilation Room B at elevation 2047 feet, 6 inches. The NRC inspector reviewed PMR 02173, dated June 8, 1987, "Scaffolding Storage Areas in RCA." This PMR included an analysis for a possible unreviewed safety question determination. The engineering disposition to using these storage areas specified that large scaffolding components must be stacked in an orderly fashion on platform trucks or preassembled scaffold sections. Small scaffold components such as base plates and clamps must be stored in metal containers which have lids. The NRC inspector noted that large scaffold pieces were stood on end and leaned against a wall. Small parts were found in piles on the floor. Containers were not found to be available in the ventilation room. The NRC inspector reviewed Procedures ADM 01-042, Revision 10, dated May 19, 1987, "Plant Modification Request Implementation," and ADM-01-057, Revision 12, "Work Request." These procedures require that a work request be used to implement any field work required by a PMR and that the work be performed as specified. The scaffold components found in apparent permanent storage but not in accordance with the PMR is considered a violation (482/8807-38). Paragraph 6 of this report contains other examples of the failure to implement procedure requirements for modification and maintenance activities.

5. Engineered Safety Features (ESF) System Walkdown (71710)

The NRC inspectors verified the operability of ESF systems by walking down selected accessible portions of the systems. The NRC inspectors verified valves and electrical circuit breakers were in the required position, power was available, and valves were locked where required. The NRC inspectors also inspected system components for damage or other conditions that could degrade system performance.

The ESF systems walked down during this inspection period and the documents utilized by the NRC inspectors during the walkdown are listed below:

System	Documents
Auxiliary Feedwater (AL)	M-12AL01(Q), Revision 0, "Piping and Instrumentation Diagram Auxiliary Feedwater System"
	Checklist CKL AL-120, Revision 10, "Auxiliary Feedwater Normal Lineup"
	STS AL-003, Revision 3, "Auxiliary Feedwater System Valve Status Verification"
Accumulator Safety Injection (EP)	M-12EP01(Q), Revision 0, "Piping and Instrumentation Diagram Accumulator Safety Injection"
	Checklist CKL EP-120, Revision 3, "Accumulator Safety Injection Lineup"
	SYS EP-200, Revision 4, "Accumulator Safety Injection Operations"
High Pressure Coolant Injection (EM)	M-12EM01(Q), Revision 0, "Piping System and Instrumentation Diagram High Pressure Coolant Injection System"
	M-12EM01(Q), Revision 2, "Piping and Instrumentation Diagram High Pressure Coolant Injection System"

Checklist CKL EM-120, Revision 6, "Safety Injection System Lineup Checklists"

M-12BN01(Q), Revision 2, Piping and Instrument Diagram Borated Refueling Water Storage System"

No violations or deviations were identified in this area of inspection.

6. Monthly Maintenance Observation (62703)

The NRC inspectors observed maintenance activities performed on safety-related systems and components to verify that these activities were conducted in accordance with approved procedures, TSs, and applicable industry codes and standards. The following elements were considered by the NRC inspectors during the observation and/or review of the maintenance activities:

- LCOs were met and, where applicable, redundant components were operable.
- Activities complied with adequate administrative controls.
- Where required, adequate, approved, and up-to-date procedures were used.
- Craftsmen were qualified to accomplish the designated task and additional technical expertise (i.e., engineering, health physics, operations) was made available when appropriate.
- Replacement parts and materials being used were properly certified.
- Required radiological controls were implemented.
- Fire prevention controls were implemented where appropriate.
- Required alignments and surveillances to verify post-maintenance operability were performed.
- Quality control hold points and/or checklists were used when appropriate and quality control personnel observed designated work activities.

Selected portions of the maintenance activities accomplished on the work requests (WR) listed below were observed and related documentation reviewed by the NRC inspectors.

No.	Activity
WR 50070-88	Pump Motor 800HP/DPAL01A semi-annual maintenance
WR 01036-88	Diesel Generator KKJ01A certain components not IEEE-323 qualified
WR 01037-88	Diesel Generator KKJ01B certain components not IEEE-323 qualified
WR 50069-88	CCW Pump motor DPEG01B 6-month oil sample
WR 52708-87	CCW pump motor DPEG01A 6-month oil sample
WR 70533-87	Erosion/Corrosion UT-remove and replace insulation
WR 01139-88	Auxiliary Feedwater Pump Motor PAL01A oil in motor appears light
WR 00218-88	Control room fan and A/C Unit B SGK04B replace handswitch

Selected NRC inspector observations for the above maintenance activities are discussed below:

During the NRC inspector's review of WR 50070-88, several problems were noted. The TS re-evaluation (Block 26) was checked off "NO", the time limit "NA" and the TS reference "NA" by the shift supervisor. As this WR was worked with the auxiliary feedwater pump out of service, Block 26 should have been checked "Yes" (the TS was applicable), a time limit of 72 hours should have been listed, and a TS reference of 3.7.1.2 should have been included. Additionally, block 18 stated "Perform Maintenance using attached Supplemental Sheets AL-17, Revision F and OS-1, Revision D." Step 2.0 of attached AL-17 states in part, "Sample the oil from each motor bearing . . . Forward oil sample to lubrication engineer for analysis. Record general condition of the old oil on the work request." The general condition of the old oil was not recorded on the work request. Administrative Procedure ADM 01-057, Revision 12, "Work Request," establishes the use of work requests, requires that the work be performed as specified, and allows block 26 to be checked "NO" only if the maintenance activity does not affect plant operation. The above examples of failure to implement administrative procedure requirements in the use of work requests are a violation (482/8807-38). The problem in paragraph 4 with implementation of modification work is another example of the same violation.

7. Monthly Surveillance Observation (61726)

The NRC inspectors observed selected portions of the performance of surveillance testing and/or reviewed completed surveillance test procedures to verify that surveillance activities were performed in accordance with TS requirements and administrative procedures. The NRC inspectors considered the following elements while inspecting surveillance activities:

- Testing was being accomplished by qualified personnel in accordance with an approved procedure.
- The surveillance procedure conformed to TS requirements.
- Required test instrumentation was calibrated.
- TS limiting conditions for operation (LCO) were satisfied.
- Test data was accurate and complete. Where appropriate, the NRC inspectors performed independent calculations of selected test data to verify their accuracy.
- The performance of the surveillance procedure conformed to applicable administrative procedures.
- The surveillance was performed within the required frequency and the test results met the required limits.

Surveillances witnessed and/or reviewed by the NRC inspectors are listed below:

- STS BG-001, Revision 3, "Boron Injection Flow Path Verification," performed January 24 and 27, and February 27, 1988
- STS BG-003, Revision 3, "Boration System Flow Rate Verification," performed April 20, 1985; October 17, 1986; and October 1, 1987
- STS RE-010, Revision 1, "RCS R Calculation," performed on January 7, 1988
- STS RE-011, Revision 2, "RCS Total Flow Rate Measurement," performed on January 12, 1988
- STS EP-200, Revision 4 "Accumulator Safety Injection," performed on March 13, 1988

Selected NRC inspector observations for the above surveillance tests are discussed below:

The NRC inspector's review of STS RE-011 identified that when the data was taken, the nuclear instruments (NIs) were reading slightly less than actual reactor power. Actual reactor thermal output was 3384 MW which is 99.2 percent of full power. Computer point REU 1169 power range nuclear channel average flux was reading 99.1 percent of full power. For conservatism the NIs should always be equal to or slightly greater than actual reactor power. The shift supervisor and a reactor engineer were informed. The NRC inspector then verified that the NIs were correctly reading slightly above actual reactor power as required and that the surveillance which adjusts the NIs had apparently set the NIs at equal to or greater than actual reactor power as required.

8. Onsite Event Followup (92700)

The NRC inspectors performed onsite followup of nonemergency events that occurred during this report period. The NRC inspectors reviewed control room logs and discussed the events with cognizant personnel. The NRC inspectors verified the licensee had responded to the events in accordance with procedures and had notified the NRC and other agencies as required in a timely fashion. The events that occurred during this report period are listed in the table below:

Date	Event	Plant Status	Cause
02/15/88	Control room ventilation isolation (CRVIS)	Mode 3	Spike on GK AI-3
02/23/88	Loss of 67 annunciators	Mode 1 (100 percent)	Blown fuse in power supply
03/17/88	Non-qualified packing box assembly	Mode 1 (100 percent)	Unknown at this time
03/26/88	CRVIS	Mode 1 (100 percent)	Chlorine Monitor, paper tape broken

Selected NRC inspector observations for the above events are discussed below:

On March 17, 1988, the licensee informed the NRC resident inspector of the discovery that a valve failed to meet certain requirements.
Valve BB PCV-455B (pressurizer spray valve) was built and installed in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code, Section III, Class 1. During the last refueling outage, which ended January 1988, the valve's packing box assembly was replaced. The licensee ordered two replacement packing box assemblies from Westinghouse who ordered them from Fisher Control. Westinghouse inadvertently ordered the packing box assemblies at a lower quality level than they should have. The replacement packing box assemblies should have been ASME Section III, SA-182-F316 material, with examination requirements as stated in Section III, certified Material Test Reports and a Code Data Report (Form N-2). Due to the error in ordering, however, the parts were procured as "commercial grade." These commercial grade parts deviated from requirements as follows:

- The material was ASTM A-276 Type 316 stainless steel bar instead of ASME SA-182-316 forging.

- No non-destructive examinations were performed as required.

- Pipe nipple material utilized for the leakoff connection was not known to be in compliance with the material requirements.

- ASME Code N-2 data reports were not supplied.

Although all other aspects of the valve were in compliance with the code, the replacement packing box assembly was not. The extra replacement packing box assembly was stored in the licensee's warehouse for future use. On March 18, 1988, the licensee has requested relief from NRC on certain code requirements in accordance with 10 CFR 50.55a(g)(6)(i).

The licensee dispatched supplier quality representatives to Westinghouse and Fisher Valve. The licensee determined that:

- The material meets ASTM A-276.

- The bar was not drawn, but was made from a billet.

- The replacement packing box assemblies were machined using the same drawing used by Fisher's nuclear facility.

- Welding of the pipe nipple to the packing box was performed using welding procedures qualified in accordance with ASME Section IX.

- Welding was performed by a welder who had been qualified to ASME Section IX but who was not up to date in qualifications.

Testing of the spare replacement packing box assembly was performed in a quality approved testing laboratory. This testing indicated the following:

- Charpy V-notch tests indicate a fracture toughness similar to that of SA-182 material.

- Chemicals analysis and mechanical tests performed parallel and transverse to the longitudinal axis of the original bar stock indicate that the actual material properties of the two replacement packing boxes, including the one currently in service, exceed the minimum specified mechanical properties and meets the specified chemical properties of SA-182 F316, with the exception of the transverse elongation (44 percent actual versus 45 percent minimum specified) which is deemed inconsequential.

The licensee also determined that the spare packing box assembly was from the same heat and same piece of material as the packing box assembly installed in the plant.

The licensee has committed to replace the subject part during the next refueling outage.

Based on the above information, the NRC Office of Nuclear Reactor Regulation has verbally granted ASME Code relief to the licensee with formal documentation to follow.

Additional inspection of this event is documented in NRC Inspection Report 50-482/88-15.

9. Radiological Protection (71709)

By performing the following activities, the NRC inspectors verified that radiologically related activities were controlled in accordance with the licensee's procedures and regulatory requirements:

- Reviewed documents such as active radiation work permits and the health physics shift turnover log.
- Observed personnel activities in the radiologically controlled area (RCA) such as use of required dosimetry equipment, "frisking out" of the RCA, and wearing of appropriate anti-contamination clothing, where required.
- Inspected postings of radiation and contaminated areas.
- Discussed activities with radiation workers and health physics supervisors.

No violations or deviations were identified in this area of inspection.

10. Physical Security (71881)

The NRC inspectors verified that the facility physical security plan (PSP) is being complied with by direct observation of licensee facilities and security personnel.

The NRC inspectors by observation of randomly selected activities verified that search equipment is operable, that the protected area barriers and vital area barriers are well maintained, that access control procedures are followed and that appropriate compensatory measures are followed when equipment is inoperable.

On March 2, 1988, the NRC inspector observed potential problems with control of a licensee designated vehicle (LDV) inside the protected area fence. The particulars of the problems were given to the licensee, who took prompt corrective action. This event was referred to NRC Region IV security specialists for followup during their next inspection.

No violations or deviations were identified in this area of the inspection.

11. Exit Meeting (30703)

The NRC inspectors met with licensee personnel to discuss the scope and findings of this inspection on March 30, 1988.