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Division of Reactor Inspection and Safeguards

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

Inspection At: Wolf Creek Nuclear Generating Station

Inspection Conducted: November 2-13, 1987 and November 9-20, 1987.

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SAFETY SYSTEMS OUTAGE MODIFICATIONS INSPECTION
at Wolf Creek Nuclear Generating Station
(Inspection Report 50-482/87032)

1.0 INTRODUCTION

The purpose of the Safety Systems Outage Modifications Inspection (SSOMI) program is to examine the adequacy of licensee management and control of modifications performed during major plant outages. The program objective is to identify strengths and weaknesses in licensee modification programs and their implementation. During this inspection, the SSOMI team examined, on a sampling basis, modifications and maintenance work activities accomplished recently at the Wolf Creek Generating Station (WCGS).

This inspection was conducted in two phases. The first phase, performed November 2-13, 1987, reviewed the design and procurement areas, and involved verification that detailed design, engineering support, and procurement activities were adequate to support the safety-related modifications performed during this outage. The second phase, conducted November 9-20, 1987, involved the review of the installation and testing of modifications performed during the outage and verification that repaired or modified components and systems had been properly installed and had been tested to ensure that they were capable of performing their intended functions. The second phase of this inspection also involved the review of selected maintenance activities.

2.0 DESIGN AND PROCUREMENT INSPECTION

2.1 Design Inspection

2.1.1 Scope

The design part of the inspection consisted of a detailed review of plant modifications and associated procedures, work requests, instructions, and drawings. The purpose of the inspection was to verify that the regulatory requirements and the design bases as documented in the Updated Safety Analysis Report (USAR) and reviewed in NRC Safety Evaluation Reports (SERs) were implemented, that correct design information had been provided to the responsible design organizations, and that the design controls applicable to the original design were utilized.

2.1.2 Discussion

The following concerns and weaknesses were identified with the Plant Modification Requests (PMR) packages reviewed:

2.1.2.1 PMR 2024: Battery Charger AC Alarm Setpoint

This PMR changed the battery charger AC input alarm setpoints to eliminate spurious AC alarms and was required because of previous changes to the AC system made in 1985 by PMR 1345 which reduced the AC input voltage at selected safety-related and nonsafety-related battery chargers. PMR 2024 indicated that the cause of spurious AC undervoltage alarms was an incorrect voltage transformation ratio assumed in the original System Relay Setting Calculation (H-12). A review of the latest revision to this calculation indicated that the transformation ratio was corrected in April 1983. The correct voltage transformation ratios were indicated for nine of the eleven battery chargers on Relay

Setting Tabulation Drawing E-11028(Q) issued in June 1984, however the new relay settings had not been added at that time.

The SSOMI team reviewed Bechtel Voltage Calculation (B-8), which was the basis of PMR 1345, searching for other possible reasons for the spurious undervoltage alarms. 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 impedence value did not agree with the transformer's nameplate data. Engineering Department personnel indicated that although the test data used in the calculation was not specific to the equipment installed at Wolf Creek Generating Station (WCGS), the error accounted for a calculated voltage error of less than two percent. The SSOMI team considered that the failure to identify the error in the Bechtel Voltage Calculation is indicative of a weakness in engineering evalutions.

2.1.2.2 PMR 899: Accumulator Level Transmitters

This PMR replaced Accumulator Tank Barton level transmitters with Rosemount level transmitters and relocated them to the side of the tank using the existing sensor taps. The SSOMI team identified the following discrepancies with this PMR:

a. The new sensor connections to the reference leg were five feet higher, which provided an additional 41.36 cubic feet of water in the tank at the minimum level setpoint than designed. As a result, 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 LOCA. The Technical Specifications require that a minimum of 818 cubic feet of water be injected from the tank into the primary system in the event of a LOCA. The actual tank pressure required with the new volume of water was not determined. The design nitrogen gas pressure of 585 psig had some allowance for conservatism, however this allowance was also unknown.

b. A root cause analysis for changing the level transmitters was not provided. Although a significant amount of data existed to justify the change, this data had not been formally communicated within the company. In addition, the Q-list, which lists all safety-related equipment in the plant, was not revised as required.

The failure to evaluate the change in nitrogen gas pressure required for Accumulator Tank injection, including calculation of the root cause for the modification is a weakness in the engineering area.

2.1.2.3 PMR 2167: Electrical Equipment Room No. 1403 Chiller

This PMR added additional cooling to Electrical Equipment Room No. 1403 and installed a room temperature indicator controller and an automatic chiller water control valve, TIC 185 and TV 185, respectively. A Field Change Request (FCR) was subsequently issued to delete TIC 185 and TV 185 and to add a manual globe valve, V150. The change required the operator to manually control the room temperature to 75°F ± 5°F by adjusting valve V150. 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. The failure to evaluate the effect of the FCR change on the PMR and the failure to document the basis for the design change is an example of a general weakness in the engineering area.

2.1.2.4 PMR 1634: Reactor Coolant Drain Tank Isolation Valve

This PMR installed an isolation valve upstream of relief valve HB-7160 to simplify inspection and repair of the relief valve. Previously, repair or replacement of the relief valve required a plant shutdown. The following discrepancies were identified:

a. The Results Engineering Group issued Temporary Modification 86-24-4B in March 1986, to gag Reactor Coolant Drain Tank relief valve HBV-7160. The 10 CFR 50.59 Safety Evaluation performed indicated that this modification did not affect the tank's overpressure protection because the tank was protected by relief valve HBV-7169. Although valve HBV-7169 had a larger spring to accommodate a higher set pressure, the evaluation indicated that the setpoint was below the design pressure of the relief tank and therefore provided adequate overpressure protection. However, the Safety Evaluation did not evaluate the required flow rate, the relative flow rates of the two valves at 110% of the tank design pressure (110 psi) and the differences in configuration. Therefore, the evaluation did not demonstrate that the second relief valve provided equivalent or adequate protection for the tank.

b. 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.

c. In addition, instrumentation was not installed at the isolation valve location to enable appropriate emergency actions if the tank was overpressurized.

The failure of the safety evaluation to demonstrate that the second relief valve provided adequate overpressure protection for the drain tank and to verify that the isolation valve would not reduce the capacity of the installed relief valve is an example of a general weakness in the engineering area.

2.1.2.5 PMR 1613: Valve Leakoff Configurations

This PMR redesigned the leakoff configuration for valves BG-HV8146, BB-8074 A,B,C, and D, BB-8055, BG-LCV459 and BG-LCV460. The PMR added flexible hoses and a shutoff valve on the leakoff lines of each of these valves. Since these valves were subject to Reactor Coolant System (RCS) pressure and temperature, the flexible hoses could have been pressurized if the valve packing leaked with the leakoff isolation valves closed. The manufacturer's rated pressure for the flexible hoses is less than RCS pressure, therefore the flexible hoses are subject to failure. The failure to adequately evaluate this modification is an example of a general weakness in the engineering area.

2.1.2.6 PMR 2109: Feedwater Valve Replacement

This PMR replaced leaking Feedwater System valves with valves which were capable of backseating with an internal diaphragm. In reviewing this PMR, the SSOMI team observed that a liquid sealant (Furmanite) was injected into the original valves to stop the valve leakage and that the use of liquid sealant injection for valve repair had been extended to safety-related valves in the primary coolant system. This practice is discussed in Section 3.2.2.2 of this report. There were no concerns identified during the review of this PMR.

2.1.2.7 PMR 2206: Auxiliary Building Fire Detection System

This PMR revised the fire detection system in the Auxiliary Building and installed 3-hour fire resistant material on a hatch. The revisions were issued in response to additional combustible material loading as a result of the installation of a tool storage area and an anti-contamination clothing storage area in the basement of the Auxiliary Building. The increased combustible loading deviates from the commitments of the USAR, as indicated below.

a. USAR Page 9.4-2, indicated that Fixed Water Suppression Systems were installed in areas with a high fire or loss potential. No fixed system was installed in the areas in question.

b. USAR Table 9.5.1-2 stated that an automatic pre-action sprinkler system was installed to protect cable trays in the Auxiliary Building at elevation 1974'-0" and that vertical cable chases were protected with an automatic wet pipe system. Uncovered cable trays containing the power cables for the "A" and "B" Auxiliary Feed Pumps pass vertically through one of the areas in question with less than 20 feet separation between the combustible material and no sprinkler system provided.

c. USAR Table 9.5A-1 indicates that safety-related systems are isolated or separated from combustible materials; the USAR also indicates that these systems are separated when practical. The installation of the two storage areas in close proximity to safety-related systems violates this guideline.

The SSOMI team was concerned that combustible loading in the Auxiliary Building was increased without implementation of the commitments of the USAR.

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 (CFC) SGN01B. The failure damaged the cooling unit and required entering a TS Limiting Condition for Operation (LCO) Action Statement. The cause of the failure was determined to be loosening of the nut which held the fan blade to the hub of the fan rotor. Subsequent inspection revealed four other loose nuts on CFC SGN01B. In addition, the licensee determined that a similar failure which was attributed to loosening of the blade nuts had previously occurred at another facility. The licensee instituted an inspection procedure to verify the tightness of the nuts during each refueling outage, but did not investigate and determine the root cause of the failure. Possible root causes for these failures are insufficient torque on the nuts to produce the required preload, excessive torque causing overstressing of the blade shafts,

inadequate blade shaft size or strength and excessive vibration. The failure to investigate the root cause of these failures is an example of a general weakness in engineering evaluations.

2.1.2.9 Appendix J Leak Test Requirements

PMR's 1143 and 2109 required the encapsulation of hinge pins for feedwater check valves and replacement of feedwater valves due to leakage. The SSOMI team determined that 10 CFR 50, Appendix J required leak testing had not been performed on these valves. The licensee indicated that the Steam Generators and attached secondary systems inside containment were closed systems and that for all accident conditions, the secondary pressure was higher than containment pressure. Therefore leak testing of the valves was not required. The SSOMI team questioned whether the Steam Generator System was required to be maintained pressurized in the event of a design basis accident and whether the instruments, instrument lines and appurtenances on the Steam Generators and other secondary systems were adequately protected in the event of a high energy line break. Further 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 general concern has been previously identified at other plants and is under review by NRR. There were no further concerns identified with respect to these PMRs.

2.1.2.10 Battery Discharge of October 15, 1987

During maintenance of the Division B Vital Bus NB02 on October 15, 1987, both Station Batteries were subjected to a deep discharge resulting in a loss of both vital buses. The root causes of the discharge are being reviewed by Region IV and will be addressed in separate correspondence. As part of this inspection, the SSOMI team reviewed the adequacy of the battery capacity and associated design calculations.

The maintenance which deenergized the bus was expected to last less than 30 hours. Based upon the nominal ampere-hour capacity of battery NK14, Operations personnel performed a calculation and estimated that the battery would provide 50 hours of service with a 35 ampere load. The SSOMI team found that the basis for the estimate that the batteries would provide 50 hours of service was inadequate. The licensee indicated that the estimate was made by dividing the battery rating of 1650 ampere-hours by 50 hours which yielded a permissible discharge rate of 35 amps. However, the actual load on the battery during the discharge was recorded in the control room each shift as 70 amperes. Battery Sizing Calculation E3, indicates that steady state loads of 220 and 100 amperes would result in discharge rates of 20 amperes per positive plate (APPP) and 16 APPP for Batteries NK12 and NK14 respectively. The battery cell characteristic curve indicates that the battery capacity would provide 10-hours of service at these discharge rates. Based on Calculation E-3 and the cell characteristic curve, the SSOMI team calculated that a 70 ampere load would have resulted in a discharge time of 35 hours.

In addition, the SSOMI team noted that during removal of the NB02 Vital Bus from service, system operating procedures SYS NB331/4 and SYS NB331/5 were not utilized. These procedures specified the requirements and precautions for

system operation and isolation and specified a maximum discharge time of 200 minutes. The licensee subsequently indicated that operational procedures are not routinely used for removing and returning equipment to service. The failure to utilize the appropriate operational procedures and incorporate the precautions and requirements of these procedures for the removal and return of equipment from service in accordance with the requirements of the Technical Specifications and 10 CFR 50, Appendix A, is considered to be a weakness. The SSOMI team considers that, if the Operations Department had complied with the procedural requirements, or had used the battery sizing calculations provided by Engineering, or had consulted with Engineering, the deep discharge and resulting loss of both vital buses would not have occurred.

The battery discharges occurred on October 15, 1987. Following these events, the batteries' conditions were not recorded prior to recharging. Based upon the results of the earlier October 7, 1987, performance discharge test on battery NK12 discussed in Section 2.1 2.12 of this report, and the fact that the unplanned discharges went deeper than the performance test, at least Cell 32, and probably other cells, reversed polarity during the unplanned discharges.

The failure to provide adequate controls for the removal from service of a safety system which resulted in a loss of the Station Batteries is a weakness. In addition, the failure to involve the Engineering Department, either through the use of system procedures or by consultation during the battery discharges, is a significant weakness.

2.1.2.11 Battery Sizing Calculation E-3

Bechtel Calculation E-3, "Class 1E Battery System," Rev. 0, dated January 12, 1987, was reviewed by the SSOMI team and the following discrepancies were identified:

a. The calculation assumed a constant load on the DC System based upon a full load rating of the 7.5 KVA for the Battery Inverter. However, in calculating the DC current, the DC voltage was taken at the nominal 125 VDC level. Because inverters try to deliver a constant KVA load, as the battery voltage decreases during a discharge the current drawn by the inverter from the battery will increase. These errors would add approximately 10% to 15% to the required battery size.

b. The input data did not document the cell characteristics used by the computerized calculation.

c. The calculation did not consider the minimum cell temperature permitted by the Technical Specification.

The battery sizes selected for batteries NK11 and NK13 are 25% and 42% larger than the calculated required size. The errors noted above can be enveloped by the existing battery and there are no safety concerns with the installed size of the batteries. The inadequacies identified in the battery sizing calculation are symptomatic of a general weakness noted by the team in engineering calculations.

2.1.2.12 Battery Performance Test

In order to evaluate the battery discharge of October 15, 1987, the SSOMI team reviewed the performance test performed for Battery NK12. The purpose of the performance test was to demonstrate the actual capacity of the battery compared to the manufacturer's published rating.

Performance Test STS MT 022, performed on Battery NK12 on October 7, 1987, was interrupted prior to the completion of the test to jumper out Battery Cell No. 32. This cell had dropped 80 millivolts in 10 minutes to 1.692 volts. The SSOMI team estimated that the battery would have reached its test limit in less than 1/2 hour if the cell had not been jumpered. The test was restarted approximately 6 1/2 hours later. When the test was restarted the battery voltage had increased to a voltage which existed two hours before the test had been stopped. As a result of the failure to recognize that the battery would recover lost capacity during the extended outage, the performance test incorrectly showed that the battery had a capacity of 138.75%. The SSOMI team calculated that if the performance test had been left to continue to completion, the battery capacity would have been calculated at 125%. While the battery has less capacity than that assumed by the licensee as a result of the performance test, the battery has 50% more capacity than that required by the test acceptance criteria. The failure to adequately test and evaluate the results of the battery performance testing is considered to be indicative of a general weakness in engineering evaluations.

2.1.2.13 DC System Low Voltage Alarms

In order to evaluate the battery discharge of October 15, 1987, the SSOMI team reviewed the alarms associated with the DC System to ensure that the design of the alarm setpoints were adequate. A review of the DC System alarms detailed on Relay Setting Drawing E-11028(Q) indicated that four DC undervoltage alarms existed in each Battery System. These alarms should have provided sufficient warning to prevent the DC System low voltages experienced on October 15, 1987.

2.1.2.14 Diesel Generator Breaker Operation

The SSOMI team reviewed the design of the closing circuit for the Emergency Diesel Generator (EDG) output breakers, NB0111 and NB0211, and concluded that the design is inadequate, in that the "anti-pumping" logic prevents the breakers from closing onto a cleared, deenergized bus. In the event that both the normal and alternate 4160 VAC Vital Bus feeder breakers, NB0109 and NB0112, are open and the EDG mode switch is in the "auto" position, the "anti-pumping" relay will be energized continuously, thus preventing the EDG output breaker from closing. Operator action is required to cycle the EDG mode switch at the local station in order to clear the "anti-pumping" logic and close the EDG output breaker to reenergize the 4160 VAC Vital Bus.

The requirement for manual operator action appears to represent an unanalyzed condition, in that the deenergized 4160 VAC Vital Bus cannot be energized automatically by the EDG in this condition. This design deficiency was discussed in a conference call between the licensee and NRC Region IV management on December 8, 1987, and will be reviewed by NRC Region IV. The inability of the EDG output breaker to automatically close onto a cleared, denergized bus is considered to be a design weakness.

2.1.3 Conclusions

For the specific packages reviewed, the SSOMI team found that modified designs were traceable to the original design bases and regulatory requirements, that correct design information was utilized, and that applicable design controls were implemented. However, the team also concluded that a weakness exists in the licensee's engineering of modifications and major maintenance activities. Examples include: the incorrect evaluation of the time of discharge of Battery NK12 prior to the loss of Vital Bus NB02, failure to prepare for the loss by providing alternate supplies, failure to provide adequate overpressure protection for the reactor coolant drain tank, failure perform adequate root cause analysis for the failure of Containment Fan Cooler fan blades, numerous errors in engineering calculations such as the Bechtel Voltage Calculation B-8 and Battery Sizing Calculation E-3, inadequate engineering evaluations of the change to the Accumulator Tank level instruments, the installation of a manual valve without temperature indication, the installation of leakoff hoses which may be overpressurized by RCS pressure, and the inadequate design of the diesel generator output breakers.

In addition, the team identified that the licensee's Operations Department failed to recognize the limitations of the Station Batteries and did not utilize operational procedures for the deenergization of the Vital Buses, which effectively precluded an engineering evaluation of the maintenance activity.

2.2 Procurement Inspection

2.2.1 Scope

The purpose of the procurement part of the inspection was to determine whether services and products acquired to support modifications made during the outage are in accordance with the licensee's commitments and regulatory requirements. The team conducted the procurement inspection at the Wolf Creek site by reviewing the procurements associated with a number of Plant Modification Requests (PMRs) for the current refueling outage, including purchase requisitions (PRs) and purchase orders (POs) used to procure components to accomplish the PMRs. The team also evaluated the implementation of the licensee's quality assurance program requirements relative to the procurement, receipt, storage and issue of components for selected PMRs, including the review of audits and surveillances conducted by the licensee's Quality Assurance (QA) and Quality Control (QC) staff in the areas of vendor surveillances and maintenance of the material storage facilities.

2.2.2 Discussion

2.2.2.1 Procedure Review

Components were selected from the PMRs and the related purchase documents for review. In addition, the SSOMI team reviewed 18 plant procedures in the procurement, administrative, and engineering areas to verify that appropriate requirements relating to the acquisition and control of equipment and materials are specified.

The SSOMI team also reviewed 50 PRs and POs for components specified in the PMRs and determined that the POs reflected the technical and quality requirements specified in the original design specifications. For the purchase of modified components, the PRs contained all the necessary information to specify the new component. The control of revisions to the component purchase specifications were determined to be adequate, in that the specifications were properly evaluated, dispositioned and documented by engineering. Where appropriate, each PO specified mandatory hold points, which required a licensee representative to witness an in-process inspection during the manufacture of the component at the vendor's facility. The procurement procedures were adequately implemented. No concerns were identified in this area.

2.2.2.2 Audit and Surveillance Review

The SSOMI team reviewed five surveillances performed on calibration activities, fourteen surveillances and one audit to verify adequate implementation of PMRs, three audits and one surveillance on the warehouse control of limited shelf-life items, and eight surveillances on the vendor's work plans performed on-site. These audits and surviellances were performed by the Quality Assurance and Quality Control Departments from 1986 to 1987. Corrective actions were promptly taken to correct adverse findings, and followup audits to verify corrective actions were complete and acceptable. No concerns were identified in this area.

2.2.2.3 Material Storage

The SSOMI team reviewed the storage of procured materials and determined that the components were adequately stored in the designated Level A, B and C areas in the warehouse. Safety-related material was stored separately from nonsafety-related material. POs for Raychem splice kits; Valcor valves including springs, modification kits and lubricants; ASCO solenoid valves; and Texas bolts were selected and storage of these items in the warehouse was observed. The components were readily traceable through the Receipt Inspection Reports to a designated storage area. The selected components had identification tags, and the necessary documentation was readily available. The warehouse was clean and measures to control the temperature and rodents were in existence. One Reactor Coolant Pump motor was in storage and the space heaters were observed to be energized. The inspectors verified the issuance and control of Raychem splice kits by matching the lot numbers used in the field with the receipt inspection documentation available in the warehouse. Eight Work Requests (WRs)for the modification of nine valves were examined to ascertain if the materials utilized in the work were traceable to documentation in the warehouse. No concerns were identified in this area.

2.2.2.4 PMR and WR Implementation

The SSOMI team reviewed Nonconformance Reports (NCRs) M-796 and M-797 dated March 30, 1987. The NCRs identified that several Limitorque motor operators had been received with improperly qualified components. The procurement documents had not specified compliance with 10 CFR 50.49 requirements. Individual parts in the motor operators were examined to determine whether they were environmentally qualified. Questionable components had been appropriately downgraded and assigned to be used in mild environments.

The SSOMI team observed the implementation of PMR 1979 and Work Request 4421-87, which required the replacement of the existing cable on ITT 763 type steam pressure transmitter AB PT 0535 on the C Steam Generator. The inspectors determined that the QC inspectors adequately verified the witness points.

The SSOMI team observed the work performed on WRs 4254, 4252 and 4255. The WRs required the inspection of flexible hoses in the vicinity of the Feedwater Isolation Valve to determine whether the flexible hoses pulled out of the fittings and to inspect the cables for damage. No damage was observed.

The SSOMI team observed the implementation of PMR 2319 and WR 4402-87, which relocated terminal boxes TB 23210, TB 23211 and TB 23212 above the flood level inside containment. One QC inspector was present to witness the work performed. Raychem splices which were used in the performance of the modification were traced to Purchase Order (PO) 522796. The receipt inspection of these splice kits was documented in Receipt Inspection Report (RIR) 510295 dated November 3, 1987. The RIR indicated that the kits were inspected and no adverse findings were identified. The certificate of conformance was attached as required.

2.2.3 Conclusions

Based on the results of this inspection, the SSOMI team concluded that the licensee has exhibited good performance in the areas of acquisition and control of equipment and materials. The team found that appropriate administrative measures are in place to implement the procedural requirements, that material has been acquired, stored and maintained in an acceptable manner, that audit and surveillance programs have been properly implemented, and that corrective actions have been effective.

3.0 INSTALLATION AND TESTING INSPECTION

3.1 Electrical and Instrumentation Modifications Inspection

3.1.1 Scope

The electrical and instrumentation portion of the installation and testing inspection consisted of a detailed review of plant modifications and associated procedures, work requests, instructions, and drawings; field verification of equipment modifications to verify conformance to design requirements and Updated Safety Analysis Report (USAR) commitments. A total of thirteen Electrical and Instrumentation and Control (I&C) Plant Modification Request (PMR) packages and five additional Safety Evaluations (SEs) were reviewed to varying degrees during the inspection. The PMRs were reviewed for adherence to regulatory requirements and procedural requirements, and installations were inspected for adherence to design and as-built requirements.

The modifications available for review were limited by the limited scope of the outage. Since significant post-modification functional testing was not performed during the inspection, completed functional tests were reviewed on a limited basis. Two Technical Specification (TS) Surveillance Tests were also observed.

3.1.2 Discussion

The following strengths and weaknesses were identified as a result of this inspection.

3.1.2.1 PMR 2262: Marathon Terminal Block Replacement

This PMR replaced Marathon terminal blocks with Raychem splices in the position indication circuits for the Pressurizer Power Operated Relief Valves (PORVs) and Code Safety Valves. Although environmentally qualified, the terminal blocks were scheduled to be replaced to eliminate any possibility of leakage currents and to assure compliance with USAR commitments for an environmentally qualified installation. The PMR package and the Work Requests (WRs) and Work Instructions (WIs) for the both PORVs and the Safety Valves were reviewed. Because installation had not been completed by the end of the inspection period, installation and testing was not inspected by the SSOMI team. There were no concerns identified in this area.

3.1.2.2 PMR 1722: Valve Motor-Operator Testing

This PMR performed valve motor-operator testing and torque and limit switch settings in response to NRC Inspection and Enforcement (I&E) Bulletin 85-03, regarding torque and limit switch settings in valve motor-operators. In addition, internal operator wiring was modified to distribute the limit switch and indication contacts to different rotors.

The PMR, PMR changes, drawings, calculations, and WR packages for several affected valves were reviewed. Operators for the five motor-operated valves (MOVs) identified below were inspected for conformance to design requirements and proper workmanship.

Valve Number	Function
BB-HV8000A	Pressurizer Block Valve
BB-HV8000B	Pressurizer Block Valve
AL-HV031	ESW to Motor Driven AFW Pump A
AL-HV030	ESW to Motor Driven AFW Pump B
EM-HV8923B	SI Pump B Suction Isolation

The following discrepancies were identified:

a. Deficiencies were identified in two of the valve operators. The spare conductors of the control cable in each Pressurizer block valve, which are located inside containment, were not properly protected to prevent possible interference with the operation of the valve. Bechtel Detail Drawing E-11013, required spare conductors of safety-related cables in the Reactor Building to be protected with Raychem end caps. Contrary to this requirement, the spare conductor in Valve BB-HV8000A was unprotected, and the spare conductor in Valve BB-HV8000B had been originally protected with tape which subsequently became unraveled. Drawing E-11013 allows the use of tape for protection of conductors outside, but not inside, the Reactor Building. In addition, insulation damage was noted on several conductors of the control cable in Valve BB-HV8000A. Based on these findings, WR 4954-87 was written for Valve BB-HV8000A and WR 4953-87 was written for

Valve BB-HV8000B to correct the deficiencies and to evaluate the insulation damage.

b. A concern was identified in the complexity of drawings required to identify the wiring configuration for conduct and verification of maintenance and modifications of the motor-operatored valves (MOVs). As many as seven different drawings and wiring lists were required to fully inspect the wiring configuration of each MOV reviewed. For example, the drawings required to inspect Valve AL-HV030 included a design schematic, a vendor wiring diagram which also included a partial schematic, two field wiring lists, and three design change documents. Other valves had wiring changes required by the vendor that were specified in writing by vendor change requests, but were not indicated on the vendor wiring drawings. Differing conventions for identifying the wire termination points on wiring diagrams were also identified. In some instances, wiring diagrams identified field wire terminations, and in others the internal wire designations duplicated field wire designations.

The difficulty in using the large number of inconsistent drawings to make changes to wiring had been previously identified by the station Electrical Maintenance Department. Engineering Evaluation Request (EER) 86-EM-03 described the problem of multiple drawings and inadequate cross referencing between plant and vendor drawings. The SSOMI team was concerned that, although the EER was approved and submitted on July 18, 1986, an evaluation or disposition for the EER had not been made.

EER 87-KC-08 was written to correct a vendor wiring drawing that used the same wire numbers twice. The engineering evaluation rejected the recommended drawing change on the basis that a review of four related design drawings indicated the wiring was adequate. The EER response referenced Bechtel Specification E-01016, "Electrical As-Built Drawing Criteria," Note N, as the reason for not correcting the wiring drawing. Note N states that internal vendor wiring inconsistencies which do not affect the circuit electrically or functionally will not be incorporated into the as-built drawings. The SSOMI team considered that this resolution was inadequate because as-built wiring drawings are routinely used for maintenance and modification activities.

Except for the discrepancies in the Pressurizer Relief Valves discussed in Section 3.1.2.2.a, the workmanship observed was good. The fact that few problems have been identified in motor-operators is attributed to the diligence of the maintenance craft and engineers and is considered a strength.

3.1.2.3 PMR 1885: Hardline Splice Replacement

This PMR replaced six safety-related, environmentally qualified, hardline splices in the Reference Junction System (RJS) of the Core Exit Thermocouple System. The qualified splices located inside containment had been replaced because their qualified life expires on June 25, 1988. The original splices had a qualified life limit of one year based on the Loss of Coolant Accident (LOCA) environment. The replacement splices were of a new design which had been qualified for a ten year life. Instrumentation and Control (I&C) WR 2145-87 implemented a Westinghouse procedure to install the hardline splices with a Westinghouse supplied kit. The PMR and WR, including the Westinghouse

supplied procedure, were reviewed. The actual splicing had been completed previously, therefore only a field inspection of the externals of the finished splices was performed. Finished splices and materials were compared with the materials listed in the work package. The installation conformed to the procedural and material requirements of the PMR package to the extent inspected. Raychem tubing and fittings were of the type and kind specified and workmanship was consistent with procedural requirements and industry standards. Quality control hold and verification points were appropriate to steps requiring witness and verification.

A minor procedural inconsistency was noted, in that the note at Step 23 of the Westinghouse procedure required elongating the flexible potting connector tube to extend approximately one inch over the field cable jacket. The procedure was specific to the RJS, but generic for field cable wire which does not have a cable jacket over the conductors. The final configuration required by the procedure was not obtained and the installation was different from that shown in the Westinghouse procedure. This deviation was not documented in the WR package. The SSOMI team concluded that this observation had no operational significance because the qualification was based on the RNC shrink tubing, full coverage by the potting connector around the shrink tubing, and RTV potting compound within the connector.

The PMR package for this modification required resistance testing after the new hardline splices had been installed. The resistance tests measured the resistance of the platinum resistance temperature detectors. The test procedure and test results were adequate with acceptance criteria for all measured values. Test equipment was specified, documented, and within the calibration interval and all test results were within the acceptance criteria. No weaknesses were noted during this review of this PMR.

3.1.2.4 PMR 2018: Asco Solenoid Valve Replacement

This PMR replaced twelve safety-related, seismic Category I, environmentally qualified, Asco solenoid valves on air-operated valves in the Reactor Coolant System, Chemical and Volume Control System, Residual Heat Removal System, High Pressure Core Injection System, and Liquid Radiological Waste System. The modification was initiated as a result of shorting of electrical wiring within the solenoid housing caused by limited space within the housing, splicing of pigtail leads, and Raychem sleeving of wires with damaged insulation. The PMR replaced solenoid valves using electrical pigtails with solenoid valves using terminals.

The PMR, WRs, procedures and documentation associated with the installation were reviewed. A field inspection was conducted on three solenoid installations that had been completed at the time of the inspection. The installations were inspected against the assembly and mounting detail drawings and work instructions in WRs that implemented the PMR. Solenoid valve serial numbers were verified against material requirements for the valves that were inspected. The following concerns were identified:

a. The 10 CFR 50.59 Safety Evaluation was inadequate, in that the seismic and environmental equivalency of the solenoid valves being installed was not documented. The written evaluation described only the improvement in safety because the leads would not short out in the new model solenoids.

The safety evaluation for changes involving substitution of components would normally be expected to reference equivalency or superiority in form, fit, function, materials, mounting, and qualification, which would equate the change to a quality level at least consistent with the originally analyzed design.

b. The air supply line for the ASCO solenoid valve EJ-HCV8890B had inadequate seismic support between the solenoid valve and the polar crane wall. The supply line had approximately eight feet of rigid and hard copper tubing that did not have support. The unsupported tubing contained both an unsupported drain valve and the solenoid air isolation valve.

c. The 3/8-inch diameter stainless steel air tubing for the Asco solenoid valve serving Valve EJ-HCV8890B had one loose tubing support between the Asco solenoid valve and the air operator.

d. WR 1042-87 which replaced Asco solenoid Valve EM-HV8881, had one instance of unclear work instructions. Step 26 required recording the valve serial number without reference to which valve. Consultation with the personnel who had written the work instructions and performed the work was required to ascertain that the serial number recorded was for the valve which had been removed. In this case, the unclear instructions did not affect safety.

3 1.2.5 PMR 2329: Raychem Splices

This PMR did not require field work and was used to document the disposition of 38 deficient Raychem splices identified in WR 4443-87 that were to be dispositioned "use-as-is". The Raychem splices were not installed in accordance with the Raychem instructions in that overlaps were less than the required two inch minimum and bend radii were less than the required five times the outside diameter. The PMR had been approved based on Wyle Nuclear Environmental Qualification Test Report No. 17859-02P, Revision A. The Wyle test report qualified seven Byron and Braidwood Generating Station specific configurations and thirteen Zion Generating Station specific configurations with overlaps as little as 1/2 inch and bend radii of 1.2 times the outside diameter.

The PMR, the WR which documented the 38 splices and the Wyle test report were reviewed. In addition, the Nuclear Plant Engineering (NPE) personnel who conducted the engineering evaluation and the Instrument Maintenance (IM) personnel who had participated in the original walkdown of the splices were interviewed. The following concerns were identified:

a. 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 for the Wyle testing. Evaluation and documentation of the LOCA conditions is required to establish a basis for use of the Wyle test report at WCGS.

b. WR 4443-87 documented the recommended disposition of "use-as-is" for the 38 splices. The WR listed the splices and stated that all splices, as a group, had seal lengths of greater than 1/2 inch but less than two inches, and that the minimum bend radius was less than five times the

shrink tube outer diameter. The WR did not document the configuration of each individual splice or provide measured bend radii or overlaps. The EER disposition that accepted the splices for "use-as-is," based on the Wyle test report, stated that the splices with bend radii as little as 1.2 times the outside diameter of the splice were tested. It also stated that the tested splices had bend radii more severe than the splices identified at WCGS. However, documentation that each splice has a bend radius greater than the bend radius tested by Wyle (stated as 1.2 times the outside diameter) was not available.

The Wyle test report covered tests of twenty Raychem splices with different configurations, tubing sizes and wire types. Only one splice tested was of a configuration similar to the 38 deficient splices at WCGS. The SSOMI team considered that the confirmation of each of the deficient splices at WCGS should be documented to confirm that the results of the Wyle tests are applicable.

c. The Wyle test report documented that the Raychem tubing was bent while it was heated. The WCGS splices were performed in accordance with vendor instructions Which allowed the splices to cool before bending. Bending the cooled Raychem tubing is less conservative because of the reduced pliability of the tubing at lower temperatures.

During a previous NRC Environmental Qualification (EQ) team inspection, questions were raised regarding the lack of documentation for sizes and lengths of machine screws used in Raychem splices. The maximum working diameter of Raychem tubing would be exceeded if the machine screws used to connect the terminal lugs were too long. Following the EQ inspection, an Engineering Evaluation Request was initiated by the licensee to establish the acceptance criteria for conducting field measurements of tubing diameter to insure that maximum Raychem working diameters were not exceeded. At the time of the SSOMI inspection, WR 4943-87 and other similar WRs were in process to inspect all suspect splices. The SSOMI team concluded that the WR appeared to adequately verify and correct Raychem problems associated with exceeding the maximum working diameter, but did not address the concerns discussed above regarding the applicability of the Wyle test report. The failure to adequately document and evaluate the engineering disposition of the nonconforming splices is a weakness in the engineering area.

3.1.2.6 PMR 1828: ESW Building Cable Replacement

This PMR replaced cables routed to the Emergency Service Water (ESW) building based upon the need for additional circuits at the ESW structure and the identification of several failures in existing cables. Cable failures were discovered during the performance of surveillance testing in which circuit breakers failed to trip open on a nonsafety-related load shed signal during Safety Injection actuation. Investigation by the licensee resulted in the identification of grounded and open circuit conditions in a number of cables which had been pulled through the duct bank system from the Main Power Block to the ESW building. Subsequent evaluation of the failed cables identified damage in the form of cuts and nicks in the insulation and jackets of these cables. This damage was assumed to have occurred during the initial cable pull. Consequently, this PMR was issued to pull new cables to the ESW building.

Although the licensee performed an evaluation of the damaged cables, an additional evaluation to determine the root cause of the failures in the original safety-related cables routed to the ESW building was not performed. The SSOMI team noted that banding material had been found in a duct bank associated with some of the damaged cables and was considered by the licensee to be the cause of the failures to the cables in that duct bank. However, the root cause for cable failures in redundant trains and cables associated with other duct banks has not been determined. The SSOMI team 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.

3.1.2.7 Containment Pressure Transmitters

During plant inspections associated with other modifications, it was noted that the connection box cover for Containment Pressure Transmitter PT-934, was missing one of two screws. Maintenance or modifications were not in progress on the transmitters. A field inspection of the other pressure transmitters identified the following discrepancies:

PT-934 - Missing 1 of 2 connection box screws.

PT-935 - No discrepancies noted.

PT-936 - Missing 1 of 2 connection box screws.

PT-937 - Missing 1 of 2 connection box screws.

PT-938 - No discrepancies noted.

PT-939 - Missing 1 of 2 connection box screws.

The licensee indicated that the missing cover screws may have been left out during a Raychem splice inspection. These discrepancies indicated a lack of attention to detail in maintenance and modification activities but were considered to be minor in nature.

3.1.2.8 Safety Evaluations

In addition to reviewing the Safety Evaluations (SEs) for the previously identified PMRs, the SEs for the PMRs in the electrical area listed below were reviewed for their adequacy. Other than the concern regarding the SE for PMR 2018 discussed in Section 3.1.2.4 above, no weaknesses were identified with respect to the following documents:

SE or PMR No.	Subject
SE 87-SE-25	MSIV N_2 Accumulator Pressure Switch.
SE 87-SE-90	Area 5 Fire Detectors Alarms.
PMR KN84-057	ESF Panel Monitor.
PMR 434	Battery Output Indication
PMR 989	Accumulator Tank Level.
PMR 1762	Reactor Trip Breakers.
PMR 1887	ESW Cable Tension.

PMR 1940	Breaker Lift Limit Switch Mounting.
PMR 1953	Conax Conductor Repair.
PMR 1975	Containment Temperature Monitoring.
PMR 1997	Motor Lead Repair.
PMR 2024	Battery Charger Alarm.

3.1.2.9 Technical Specification Tests

The SSOMI team monitored the performance of Technical Specification Surveillance Tests (STS) STS IC-280A, "Analog Channel OP Test Ctrl Rm CL Detection Train A," and STS IC-433, "Channel Calibration NIS Post Accident Monitoring N61," to ensure that they were performed in accordance with the requirements of the listed test procedures and were administratively controlled by procedure ADM 02-300, "Surveillance Testing." STS IC-280A verified the operability of the Chlorine Detection Control Room Ventilation Isolation System and STS IC-433 calibrated the Post Accident Monitoring Nuclear Instrumentation System (NIS).

The performance of STS IC-280A was in accordance with requirements. Test personnel were knowledgeable and appeared qualified to accomplish test objectives. However, several weaknesses were observed in the test instructions of STS IC-433. A lack of detail in some sections of the procedure resulted in confusion on the part of the Test Technician. For example, a note to Section 6.3.1.3.1 incorrectly referenced "zero power" as a prerequisite for bypassing certain steps of the procedure. This note should have referenced a Nuclear Instrumentation System (NIS) Source Range level. In addition, Sections 6.3.2.2.2 and 6.3.2.3.2.1 specified test equipment connections which could not be accomplished without removal of the appropriate solid state circuit card. The removal and reinstallation of the circuit card and any subsequent requirement for equipment warm up were not detailed in the procedure. Consequently, interpretation on the part of test personnel was required in order to accomplish the test objectives. As a result of these discrepancies, the test performance was suspended until the procedure was revised.

The SSOMI team was unable to determine the extent of this concern because of the limited sample of tests and test procedures available for review. The test personnel were knowledgeable and this test could have been accomplished through application of the skills and experience which they possessed. However, 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.

3.1.3 Conclusions

In general, the work performed on the electrical and instrumentation modifications was acceptable and met applicable requirements. The several equipment deficiencies and minor work practice problems identified with respect to the Pressurizer Block Valves in PMR 1722 and Containment Pressure Transmitters are considered to be isolated cases.

The SSOMI team concluded that the engineering evaluations associated with the performance of plant modifications were a weakness, as evidenced by the acceptance of nonconforming Raychem splices without adequate verification in PMR 2329, inadequate root cause analysis for ESW cable failures in PMR 1828, and inadequate safety evaluation for the replacement of solenoids in PMR 2018. The

implications of the cable pull failures are of particular concern and should be evaluated further.

The SSOMI team had concerns regarding licensee management's responsiveness to identified problems. The drawing system used for maintenance to identify as-built wiring configurations was previously identified to be overly complex and, in some cases, inconsistent. Corrective action had not been implemented. Although few wiring discrepancies had been identified and the overall workmanship was good, the difficult drawing system had the potential to create problems in future work.

3.2 Mechanical Modifications and Maintenance Inspection

3.2.1 Scope

The SSOMI team reviewed Plant Modification Request (PMR) packages which were completed, in progress, or pending, for completeness and conformance to licensee procedures, the licensing basis as described in the Updated Safety Analysis Report (USAR), and national codes and standards. The packages generally contained the scoping documents for the modifications, 10 CFR 50.59 Safety Evaluations, Nuclear Plant Engineering (NPE) engineering dispositions, design input and output listings and indices, Plant Modification Package Change Notices, and Field Change Requests. The SSOMI team field inspected to the latest as-built drawings and the configurations required by the PMR Work Request (WR) packages, including use of proper materials.

Detailed WR packages associated with the PMRs were also inspected. The WRs provided the detailed work instructions, quality requirements, drawings, and testing requirements to the installation personnel. In addition, WCGS engineering and maintenance staff were interviewed regarding various aspects of PMRs, WRs and other documents.

Mechanical testing was evaluated by reviewing the test procedures, observing testing in progress and reviewing completed documentation related to testing. Test procedures were reviewed to determine whether they were clearly defined, completed, and adequately controlled. Testing was observed to evaluate the test procedure adequacy, adherence to procedure by test personnel and handling of test deviations and unexpected conditions. Documentation packages from modifications previously completed and completed Inservice Tests (ISTs) for the AFW pump were reviewed for completeness, accuracy, proper resolution of deviations, and proper reviews.

3.2.2 Discussion

3.2.2.1 Temporary Modification TMO 87-120 GK: Clamped Open CRVIS Damper

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.

TS 3/4.7.6, "Plant Systems - Control Room Emergency Ventilation System," requires that the Control Room Ventilation Isolation System (CRVIS) be operable in all modes and capable of pressurizing the control room to 0.25 inches of water (gauge) upon detection of radiation or toxic gas in the air intakes. In addition, Section 3.1 of the USAR and 10 CFR 50, Appendix A, require that safety systems be designed to automatically protect against single failures of passive and active components. The system configuration established by the temporary modification provided a bypass path for supply air to the control room if the Train A fan failed to automatically start on a CRVIS initiation signal. Upon CRVIS actuation, the Train B CRVIS fan would start and discharge air to the fan discharge plenum, however the clamped open Train A damper would permit backflow through the idle Train A CRVIS fan and bypass the control room. In this configuration, the CRVIS would not be able to provide the required positive pressure in the control room, assuming the single failure of Train A CRVIS fan. Additionally, 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.

Additional concerns associated with this temporary modification were noted as follows:

a. The licensee failed to recognize the requirement for the safety system to remain operable with a single failure without operator action. In order to ensure the operability of the CRVIS, the temporary modification required an operator to remove the clamp from the failed CRVIS damper. Although these actions would permit Train A to be isolated upon fan failure and therefore satisfy the single failure design of the system, 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.

b. 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 emergency instructions require the operator to don a self-contained breathing apparatus, go to the damper, climb a ladder to the blocked damper, remove the clamp and manually ensure that the damper has closed. Furthermore, the SSOMI inspector noted that these instructions would have been difficult to implement in an emergency because they were not found in the Alarm Response Procedure as normally expected, but were included as an addendum to the temporary modification.

c. Despite having performed repeated Safety Evaluations on this and other CRVIS dampers which were similarly clamped open, the licensee failed to recognize that additional testing or calculations were necessary to verify that the system remained operable with the temporary modification implemented.

d. Appropriate corrective action in preventing repeated damper failures was not taken. Five CRVIS damper failures were experienced during the period of June 25 to November 3, 1987. As a result, replacement operating gear to repair the broken train A CRVIS supply damper was not available. Appropriate corrective actions require a consideration of past defects and noncompliances in basic components important to safety. WCGS Proce-

dure ADM 01-033, "Instructions Describing Reportability Review and Documentation of Licensee Event Reports and Defect Deficiencies," Rev. 16, specified evaluation and reporting requirements pursuant to 10 CFR 21. The licensee had not identified the above failures as potentially reportable nor evaluated the failures per the above procedure.

3.2.2.2 PMR 2106: Pressurizer Spray Valve Bonnet Repair

This PMR involved the installation of gland plugs or set screws in four injection holes as a permanent modification to the Pressurizer Spray Valve. The four injection holes were drilled in the bonnet of the Pressurizer Spray Valve in order to provide an injection path for a liquid sealant (Furmanite) which was used to repair the body-to-bonnet steam leak.

The engineering disposition of Engineering Evaluation Request (EER) 85-XX-37, which requested that NPE approve the use of certain sealing compounds in temporary repair procedures at the discretion of the Maintenance Superintendent, previously approved the repair of leaking mechanical joints in piping components such as flanges, valve packing, and bolted valve bonnets by the use of sealant injection, provided that the sealant chemistry requirements, application procedure, and system limitations specified in the engineering disposition were followed. The disposition approved the generic use of liquid sealing compounds such as Furmanite and authorized drilling holes into pressure retaining parts of ASME Code Class I components in order to facilitate the repair process.

WR 00101-87 and associated revisions, the vendor work request and procedure, the engineering disposition to PMR 2106 and other associated documentation which were a part of the work package were reviewed. The following concerns were identified:

a. 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 Justification of Engineering Resolution for EER 85-XX-37 indicated that the disposition ensured that Code requirements were not violated. NPE subsequently indicated that the requirements cited in the Justification of Engineering Resolution were obtained from the vendor and not from the ASME Code as indicated. When requested by the SSOMI team, the licensee could not demonstrate that the requirements provided by the vendor met the Code requirements.

b. 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 I components involved changes to the facility. Although the safety evaluation performed for WR 00101-87, Rev. 3, correctly determined that the holes drilled in the pressurized bonnet were a change to the facility, the explanation given for determining whether this instance involved an unreviewed safety question was inadequate because it failed to address the safety significance of the modification to the Pressurizer Spray Valve using EER 85-XX-37.

c. The engineering disposition to EER 85-XX-37 required the sealing compound injection pressure to be calculated so as to limit the injection pressure and thereby limit the stress on the flange bolts. Documentation of these calculations was not available. Additionally, the maximum pressure used to inject the sealant compound in the body-to-bonnet area of the Pressurizer Spray Valve was not recorded. Because the maximum pressure was not calculated and records did not exist to demonstrate the pressure used, the flange bolt stresses could have been exceeded for the Pressurizer Spray Valve during the injection process.

d. 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).

3.2.2.3 PMR 2084: CCW Pipe Wall Thinning.

This PMR involved application of a weld overlay on a Component Cooling Water (CCW) line servicing the Train A Residual Heat Removal (RHR) Heat Exchanger in order to repair a pipe section downstream of Valve EJ-V033. The licensee's ultrasonic examination of the piping section determined that the pipe section was below minimum wall thickness. Although subsequent ultrasonic examinations determined that these results were erroneous because laminar inclusions (acceptable conditions), rather than the wall inner diameter, were being identified, the SSOMI team identified the following concerns:

a. 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 licensee subsequently issued a guidance memorandum on July 29, 1987, directing the plant operators to consider systems inoperable if requirements for minimum piping wall thickness were violated. The general matter of the licensee's handling of these and other similar operability matters was the subject of an NRC:NRR-licensee meeting in NRC headquarters on November 17, 1987.

b. The piping was repaired without being isolated and drained of water, even though the test results showed wall thinning down to nearly 1/16 inch (about 22 percent of nominal). Although Welding Procedures WPS1-0000, "ASME/ANSI General Requirements," Rev. 1, and WPS1-0101T30 and WPS1-0101S01, "Welding of P1 Materials," Rev. 2, permitted welding under these conditions, the SSOMI team considered that the procedure represented a high risk of wall burn through, threatened system pressure boundary integrity, and therefore represented a nonconservative repair procedure.

c. 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," Rev. 2. ADM 01-036 required that work packages contain complete and concise work instructions to accomplish repairs and provided guidance for content. Revision 2 to the WR implemented changes in the basic repair procedure and included requirements for in-service leak testing and radiography. The existing work instruction was annotated to delete all existing steps by Revision 2, but replacement work steps

were not provided. Additionally, the repairs were completed without craft or QC signoff of the revised work instructions.

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

e. 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, Rev. 1, for about six months. 10 CFR 50.55a(g)(5)(iii) requires NRC notification when conformance with certain code requirements is impractical. The licensee did not initially notify the NRC when the radiography was deferred. Following identification of similar oversights by the NRC Resident Inspectors, the licensee submitted a Code Relief Request for the radiography of PMR 2084 repairs on August 24, 1987.

3.2.2.4 PMR 2116: Valves EF-V090, EF-V058, EF-HV47 and EF-HV48 Hard Surfacing.

This PMR involved the final repairs of Essential Service Water (ESW) system piping, including EF-V090 piping and piping downstream of Valve EF-V058 on the opposite train. Additionally, this modification involved replacement of a carbon steel 24" x 16" bell reducer with a stainless steel replacement more resistant to erosion and corrosion. Adjacent piping was also hard surfaced with either stainless steel or stellite weld overlays. The PMR, associated WR packages and in-progress welding and fitup of a new piping bell reducer at EF-V058 were reviewed. On November 12, 1987, during the SSOMI inspection, work on the above WRs and all safety-related pipe fitting and welding activity conducted by the Maintenance Department were suspended by QA Work Hold Agreement #23. The Work Hold Agreement, issued jointly by the QA and Maintenance Departments, cited fourteen Conditions Adverse to Quality involving PMR 2116 as the basis for the work suspension. These included improper piping cut outs, marginal or inadequate piping fitups, failure to temporarily support piping for spool removal, out of tolerance spool fabrication, bypassed QC hold points, improper weld buildups and overlays, and others. This work hold followed a previous QC requested work stoppage.

Maintenance and QC Department management indicated that the problems associated with PMR 2216 were caused by the Maintenance Department's lack of experience in major pipefitting modifications. Previously, such work was performed by contractors under the direction of the Facilities and Modifications (F&M) Department. This modification was transferred to F&M on November 13, 1987, and work resumed following review and revision of work instructions on November 14, 1987.

No concerns were identified with respect to this work. However, 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. The testing pressurized a portion of the ESW System to 220 psig and required a test relief valve set at 239 psig. The procedure provided single valve isolation for thirteen heat exchangers served by the piping. No provisions were included to ensure that the adjacent components were aligned such that test boundary valve leakage will be vented to atmosphere or the untested portion of the system.

The SSOMI team was concerned that test boundary valve leakage without component protection could result in pressurization of the heat exchangers in excess of the pressures allowable under ASME Section XI.

3.2.2.5 PMR 1903: Repair of ESW Leakage/Wall Thinning

This PMR and its associated documentation specified interim repairs on the piping reducer downstream of Valve EF-V090 and ultrasonic examinations of other piping. The activities associated with this PMR had previously been reviewed by the NRC Resident Inspectors as discussed in Inspection Reports 50-482/87-15 and 87-20. Review of this PMR was performed as a result of concerns identified with PMRs 2084 and 2116. No additional concerns were identified.

3.2.2.6 PMR 0904: Essential Service Water Check Valves

This PMR installed isolation valves for Essential Service Water (ESW) Check Valves EF-V0046 and EF-V0076 to improve the maintainance and testing of the check valves. The PMR added two new gate valves, EF-V0345 and EF-V0346. 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." Criterion XVII requires that quality records shall, as a minimum, identify the inspector or data recorder. Other licensee procedures, such as surveillance procedures, typically include a tabulation which correlates the individual's initials and signatures to permit positive identification. The SSOMI team considered that the personnel performing valve lineups should be clearly identified.

3.2.2.7 PMR 1363: Charging Pump Control Valve Cavitation Damage

This PMR replaced the trim in Charging Pump Flow Control Valve BG-FCV121 with a new design. The former valve trim experienced cavitation damage because of the high flow and high pressure drop while in service.

WR 4430-86, used to implement PMR 1363 for Valve FCV-121, did not note that PMRs 1613 and 1635 were performed concurrently. Contrary to Note 9 on Copes Vulcan Drawing D-283137, which required a total of nine packing rings to be installed, Step 7 of the work package was annotated to indicate that twelve rings of packing were installed. The licensee indicated that the reason there were more packing rings installed was because the licensee had implemented PMRs 1613 and 1635 concurrently with PMR 1363. PMRs 1613 and 1635 removed packing leakoff piping and lantern rings and replaced braided asbestos packing with graphite packing. More rings of replacement packing had to be used to replace the originally installed packing. A review of PMRs 1613 and 1635 by the SSOMI team found that the installation of the new packing material was acceptable, and the team noted that a supplemental valve repacking instruction was attached to the WRs which implemented the requirements of PMR 1635 for packing replacement. The failure to annotate WR 4430-86 to cross-reference the above PMRs for documentation completeness is considered a weakness.

3.2.2.8 Safety Analysis Review

The SSOMI team reviewed a number of 10 CFR 50.59 Safety Evaluations which had been performed relative to plant modifications. The licensee had implemented an extensive Safety Review Checklist, which required a review of fire protection, As Low as Reasonably Achievable (ALARA) considerations, environmental qualification of equipment, and environmental protection matters, as well as 10 CFR 50.59 requirements. The team considered the review checklist to be rigorous and thorough. Procedures KPN-D-304, "Licensing Review and Safety Evaluation," Rev. 1, and ADM 01-022, "Authorization of Changes, Tests, and Experiments," Rev. 3, provided instructions for Safety Evaluation performance for NPE and the plant staff.

From a sample of 220 Safety Evaluations, the team selected 25 for detailed review. A number of minor deficiencies were identified and provided to the licensee for information and correction.

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.

Three direct acting Code safety valves made by the Crosby Valve and Gage Company were installed on the Pressurizer. ASME Section XI, Article IWV-3500, required that the valves be tested on a rotating basis so that each valve is tested at least once every five years. This five year cycle for valve testing began with facility commercial operation.

As a result of NRC Region IV concerns with the adequacy of the testing of the safety valves, the licensee conducted setpoint testing of the three installed valves (BD-8010A, BD-8010B, and BD-8010C) during the current outage. The licensee elected to test the valves by the use of a local bench test because of dissatisfaction with the contractor which had previously tested the Main Steam safety valves. The pressurizer safety valves were previously tested by the valve vendor between 1977-1978 and had not exceeded their ASME Code testing periodicity requirement. Additionally, three spare valves had been tested in March 1987. The inspector reviewed the detailed test procedures and data and identified the following concerns:

a. Five of the six valves tested had setpoints well below the minimum TS limit of 2461 psig. Additionally, the "C" Valve setpoint was found to be below the Pressurizer Power Operated Relief Valves (PORVs) setpoint, although the licensee indicated that this safety valve had not lifted during a prior plant transient which had caused the PORVs to lift.

   The valves were bench tested in accordance with a method acceptable by the ASME Code, using a low volume, high pressure test rig. The valves were variously heated to simulate ambient installed conditions using either heat bankets or a "hot box" equipped with electrical heating elements. As

corrective action the "B" and "C" valves were replaced with spare safety valves and the "A" valve was reset and reinstalled.

b. The SSOMI team also noted that the safety valves exhibited greater than expected setpoint deviation with respect to temperature variations. Diagnostic testing of the safety valves, performed as a result of discrepancies identified in the "B" and "C" valve testing, indicated that the valve setpoint dropped about 0.87 psig per degree Fahrenheit (F) increase. The diagnostic testing was performed at various ambient temperature conditions ranging from 80 to 175 degrees F and various nozzle ring settings. The nozzle ring adjusts the valve lift sensitivity. The licensee performed an informal statistical evaluation of this data and found the temperature setpoint relationship to be linear. A review of the Electric Power Research Institute (EPRI) test data for similar valves found the temperature setpoint shift observed at WCGS to be two to three times the maximum inferred from the EPRI data.

c. The temperature conditions used to test the installed valves were different from the temperature conditions used to test the spare valves. The installed valves were tested per procedure STS MT-005, "Pressurizer Code Safety Valve Operability," Revision 1, which specified that the valves be heated to 120 to 200 degrees Farenheit (F) to simulate the ambient condition as required by the TS. The spare safety valves were tested by detailed test instructions based on the Crosby Valve Manual which specified that the valves be heated to 120 to 140 degrees F. Additionally, 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" valve ambient conditions as required by TS 3/4.4.2.2.

d. 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 degrees 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.

e. Pressurizer safety valves are equipped with guide and adjusting rings which control the valve blowdown. The ASME Code and associated test requirements assume that guide ring settings established during vendor certification testing result in repeatable valve configurations which do not require periodic testing. Therefore bench testing, which does not change the ring settings, is allowed even though it does not verify actual blowdown performance. During the inspection, 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.

f. 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. Incorrect ring settings had been found to result in insufficient relief capacity and have possibly contributed to premature valve operation and/or reseating failures. Although the I&E Information Notice addressed Main Steam safety valves and not the Pressurizer safety valves, the Pressurizer safety valves at WCGS are essentially identical in configuration to the Main Steam safety valves and should have been evaluated.

3.2.3 Conclusions

Modifications such as clamping open the CRVIS suction dampers, drilling of a safety-related valve for sealant injection and the weld repair of operational Component Cooling Water piping are examples of an unconservative approach to plant modifications. The team noted a number of instances in which the licensee had performed inadequate Safety Evaluations in the mechanical areas.

The SSOMI team also identified specific instances of inadequate procedures and failure to follow procedures. These weaknesses, including the failure to update detailed work instructions for the ESW piping repairs, had the effect of causing or contributing to other implementation deficiencies such as a missed leak test for CCW piping in PMR 2084.

Although few tests were conducted during the inspection period, the review of pressurizer safety valve testing procedures and testing results revealed inconsistent testing requirements and lack of communication between the Maintenance and NPE departments. Additionally, there was a lack of management involvement in complex and difficult tests conducted by the Maintenance Department. Involvement by appropriate levels of management would have prevented many of the problems identified during this inspection.

4.0 EXIT MEETING

The design and procurement team conducted an interim exit meeting on November 13, 1987, with licensee management to provide a summary of issues identified during the design and procurement parts of the inspection. The installation and testing team leader conducted a final exit meeting on November 20, 1987, to provide a summary of issues identified during the installation and testing parts of the inspection. Specific observations were presented for each area inspected. The licensee was presented an opportunity to question the observations from both portions of the inspection. Mr. C. J. Haughney, Branch Chief, Special Inspections Branch, NRR, and Messrs. J. M. Montgomery, Deputy Regional Administrator, Region IV, and L. J. Callan, Director, Division of Reactor Projects, Region IV, represented NRC management at the final exit meeting. The scope of the inspection was discussed and the licensee was informed of the conclusions identified in the course of the inspections. Issues which were required to be resolved before restart of the unit were identified as indicated in the cover letter to this report.

5.0 BACKGROUND

5.1 Personnel Contacted

A large number of WCNOC[1] personnel were contacted during the inspection. The following is a brief list of the key personnel involved:

Name	Title
*#B. Withers	President, WCNOC
*#R. Grant	Vice-President Quality
*#F. Rhodes	Vice-President Operations
*#J. Bailey	Vice-President Engineering
*#G. Boyer	Plant Manager
*#O. Maynard	Licensing Manager
#R. Belote	Nuclear Safety Engineering Manager
*#A. Freitag	Nuclear Plant Engineering Manager
*#J. Pippin	Nuclear Plant Engineering Manager
*#R. Holloway	Facilities and Modifications Manager
*#C. Snyder	Purchasing and Material Services Manager
#C. Estes	Operations Superintendent
#A. Clason	Maintenance Services Superintendent
#R. Benedict	Quality Control Superintendent
#B. McKinney	Technical Support Superintendent
*#M. Williams	Regulatory Assurance Superintendent
M. Rich	Maintenance Superintendent
#M. Nichols	Plant Support Superintendent
D. Fehr	Licensed Training Superintendent
#W. Nelson	Administrative Services Director
#B. Goshorn	Wolf Creek Coordinator
*#W. Lindsay	Quality Systems Supervisor
*#C. Fowler	I&C Supervisor
*#D. Walch	Maintenance Engineering Supervisor
E. Creel	Nuclear Coordinator - KG & E
R. Robinson	I & E Corrdinator
D. Kowalski	Lead Engineer - NPE
B. McKinney	Superintendent - Technical Support
G. Rathbun	Management Systems Manager
C. Delons	Quality Plant Supervisor
F. Hall	QC Inspection Supervisor
T. Foster	Modification Engineering Supervisor
#W. Rudolph	Manager, Quality Assurance
C. Younie	Supervisory Operator
P. Martin	Shift Supervisor
D. Neufield	Shift Supervisor
O. Korbelek	Shift Supervisor
M. Bove'	Lead Electrical Maintenance Engineer
T. Graf	Lead Electrical Engineer
L. Courley	Lead Facilities and Modifications Engineer
T. O'Hearn	Lead Electrical Engineer, NPE -
J. North	Engineering Specialist, F&M
L. Gourley	Sr. Engineering Specialist, F&M
#R. Blecha	Mechanical Maintenance Engineer
J. Winkle	Authorized Nuclear Inspector, HSB
J. Houghten	Operations Coordinator

E. Torpey	Welding Engineer
R. Gesling	Fire Protection Specialist
M. Piteo	Engineering Specialist
R. Reitman	Results Engineer
*#C. Hoch	Quality Assurance Technician
#S. Sparks	Licensing Engineer
#K. Clair	Maintenance Engineer
*#H. Chernoff	Licensing Engineer
*#G. Pendergrass	Licensing Engineer

* Attended design and procurement inspection exit meeting on November 13, 1987.

\# Attended installation and testing inspection exit meeting on November 20, 1987.

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[1]Kansas Gas and Electric Company (KG&E), Kansas City Power and Light (KCPL), and Kansas Electric Power Cooperative, Inc. (KEPCo) are co-owners of the Wolf Creek Generating Site (WCGS). KG&E and KCPL were co-applicants for the operating license with KG&E designated as the Operating Agent responsible for operation of the facility. A separate entity, the Wolf Creek Nuclear Operating Corporation (WCNOC), was recently formed to operate the facility for its owners.

5.2 References

Bechtel Drawing E-11013(Q), "Installation Inspection & Testing Details for Electrical Equipment & Cable," Rev. 2

Kansas Gas & Electric Company, (KG&E) Nuclear Plant Engineering (NPE) Procedure, KPN-C-300, "Plant Modification Process," Rev. 4

KG&E NPE Procedure, KPN-C-301, "Initiation of Plant Modifications," Rev. 6

KG&E NPE Procedure, KPN-C-302, "Engineering Study for Plant Modification Requests," Rev. 7

KG&E NPE Procedure, KPN-C-304, "Priority 1 Modifications," Rev. 1

KG&E NPE Procedure, KPN-C-306, "Preparing the Plant Modification Request for Release," Rev. 2

KG&E NPE Procedure, KPN-C-307, "Revising the Plant Modification Request," Rev. 3

KG&E NPE Procedure, KPN-C-308, "Closeout of Completed Plant Modification Requests," Rev. 2

KG&E NPE Procedure, KPN-C-309, "Plant Modifications and Design Development," Rev 1

KG&E NPE Procedure, KPN-D-303, "Determination of Safety Classification," Rev. 2

KG&E NPE Procedure, KPN-D-304, "Licensing Review and Safety Evaluation," Rev. 1.

KG&E NPE Procedure, KPN-D-306, "Requesting Changes to the FSAR and Technical Specifications," Rev. 2

KG&E NPE Procedure, KPN-D-319, "Environmental Qualification Review of Electrical Equipment to 10 CFR 50.49," Rev. 1

KG&E NPE Procedure, KPN-E-301, "Design Interfaces," Rev. 4

KG&E NPE Procedure, KPN-E-302, "Design Input," Rev. 2

KG&E NPE Procedure, KPN-E-306, "Design Specifications," Rev. 3

KG&E NPE Procedure, KPN-E-308, "Interdisciplinary Reviews," Rev. 5

KG&E NPE Procedure, KPN-E-309, "Design Verification - Safety Related," Rev. 3

KG&E NPE Procedure KPN-E-310, "Design Verification Special Scope," Rev. 4

KG&E NPE Procedure KPN-E-311, "Design Verification - Non Safety," Rev. 4

Wolf Creek Nuclear Operating Corporation (WCNOC), Administrative Procedure, ADM-01-053, "Engineering Evaluation Request," Rev. 5

WCNOC Administrative Procedure, ADM-01-057, "Work Request," Rev. 12

WCNOC Administrative Procedure, ADM-01-108, "Outage Planning," Rev. 2

WCNOC Administrative Procedure, ADM-01-112, "Acceptance Testing Using (SPE) Procedures," Rev. 1

WCNOC Administrative Procedure, ADM-02-100, "Clearance Order Procedure," Rev. 15

WCNOC Administrative Procedure, ADM-02-0300, "Surveillance Testing," Rev. 10

WCNOC Administative Procedure, ADM-08-201, "Control of Maintenance and Modifications," Rev. 3

GLOSSARY

AC	alternating current
AFW	auxiliary feedwater
ALARA	as low as reasonably achievable
ANSI	American National Standards Institute
APPP	amperes per positive plate
ASME	American Society of Mechanical Engineers
BOP	balance of plant
CCW	Component Cooling Water
CFC	Containment Fan Cooler
CRVIS	Control Room Ventilation Isolation System
CWR	Corrective Work Request
DC	direct current
DDCN	Design Document Change Notice
EER	Engineering Evaluation Request
EG	Component Cooling Water System designation
EPRI	Electric Power Research Institute
ESF	Engineered Safety Feature
ESFAS	Engineered Safety Features Actuation System
ESW	Essential Service Water
F	Fahrenheit
F&M	Facilities and Modifications Group
FSAR	Final Safety Analysis Report
I&C	Instrumentation and Control
INPO	Institute of Nuclear Power Operations
KG&E	Kansas Gas and Electric Company
LCO	Limiting Condition for Operation
LOCA	Loss of Coolant Accident
MOV	motor-operated valve
MSIV	Main Steam Isolation Valve
NDE	nondestructive examination
NIS	Nuclear Instrumentation System
NPE	Nuclear Plant Engineering Department
NRR	NRC Office of Nuclear Reactor Regulation
P&ID	piping and instrumentation drawing
PMR	Plant Modification Request
PO	Purchase Order
PORV	Power Operated Relief Valve
PR	Purchase Request
psig	pounds per square inch gauge
PSRC	Plant Safety Review Committee
QA	quality assurance
QC	quality control
RCS	Reactor Coolant System
RHR	Residual Heat Removal System
RIR	Receipt Inspection Report
RJS	Reference Junction System
SE	safety evaluation
SER	Safety Evaluation Report
SI	Safety Injection
SSOMI	Safety Systems Outage Modifications Inspection (NRC)

STS	surveillance test, Technical Specifications related
TS	Technical Specifications
USAR	Updated Safety Analysis Report
UV	undervoltage
WCGS	Wolf Creek Generating Station
WCNOC	Wolf Creek Nuclear Operating Corporation
WI	Work Instruction
WR	Work Request