Enclosure 3

U. S. NUCLEAR REGULATORY COMMISSION
OFFICE OF NUCLEAR REACTOR REGULATION

Report No. 50-482/88-200 Docket No. 50-482 License No. NPF-42

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Facility: Wolf Creek Generating Station

Inspection At: Wolf Creek, Burlington, Kansas, June 6-17, 1988

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SUMMARY

Areas Inspected

This special, announced Nuclear Regulatory Commission (NRC) team inspection was the seventh in a series of NRC Headquarters-directed Quality Verification Function Inspections (QVFIs). The inspection was performed to assess the line organization's support and contribution to plant quality and the quality verification organization's ability to identify, solve, and prevent the occurrence of safety-significant deficiencies in the functional areas of plant operations and maintenance. Another area that was evaluated during the QVFI was the effectiveness of management in ensuring that identified quality deficiencies were responded to promptly and completely.

Results

Within the functional areas of operations and maintenance, six potential enforcement findings (PEFs) were identified: (1) six examples of not taking appropriate corrective actions to prevent recurrence of plant system and component deficiencies, (2) not having procedures and instructions appropriate for the bearing removal activities on a component cooling water pump, (3) not obtaining and performing evaluations of applicable service information letters from the emergency diesel generator vendor, (4) not verifying that four seismic and vibration control supports were installed on the emergency diesel generator turbocharger cooling water piping as specified by the vendor's design drawing, (5) not posting a fire watch after a fire barrier seal in a penetration was determined to be unqualified, and (6) not declaring a loop of the Essential Service Water System inoperable when it was determined it did not meet its specified design requirements.

In addition, two observations were identified: (1) a lack of a feedback mechanism for maintenance personnel to report problems and recommendations to procedure writers, and (2) a lack of an adequate methodology to calibrate the resistance temperature detectors for the reactor coolant system.

1 INTRODUCTION

This special, announced NRC team inspection at Wolf Creek Generating Station (WCGS) was performed to evaluate the acceptability of the line and quality verification organizations' activities and management's support of these activities. The inspection was the seventh in a series of NRC headquarters-directed inspections performed under the guidance of NRC Inspection Manual Temporary Instruction 2515/78, "Inspection of Quality Verification Functions." The inspection consists of personnel interviews, direct observation of in-process activities, and review of work documents.

Quality Verification Function Inspections (QVFIs) are not intended to verify licensee compliance to administrative controls; they are intended to verify the technical adequacy of safety-related activities. However, if deficiencies are found in these activities, the underlying procedures and administrative controls are reviewed. The intent of these inspections is to improve plant operational safety through inspection processes that are focused on activities that affect plant safety and reliability.

The QVFI at Wolf Creek focused on plant operations and maintenance of plant systems and components. The inspectors reviewed selective samples in these and closely associated areas to identify safety-significant problems to be used as the vehicles for evaluating the effectiveness of quality achievement and verification. The results of this review are discussed below and the inspectors' more significant findings are categorized as potential enforcement findings and observations.

Potential enforcement findings are apparent violations of regulatory requirements that will be further evaluated by NRC Region IV management for possible enforcement action. Observations are items that may not violate any regulatory requirements and may not violate plant procedures, but that appear to be less than optimum. Observations are being referred to NRC Region IV and NRC Headquarters Staff and may require inspectors to perform followup reviews during subsequent inspections.

2 PLANT OPERATIONS

2.1 Control Room and Operations Activities

2.1.1 Inspection Results

The NRC inspectors observed control room and other operations activities, interviewed control room personnel, and reviewed pertinent documents related to operations activities. The inspectors observed control room decorum, control room shift turnover during dayshift and backshift, main turbine valve cycling, maintenance and testing of the reactor trip breakers, a walkdown of the auxiliary feedwater system, and a transient involving a loss of automatic feedwater control and the subsequent recovery in the control room. The team inspected other plant areas to verify operability of equipment, control of ignition sources and combustible materials, proper condition of fire detection and extinguishing equipment, adequacy of maintenance activities, and adequacy of selected surveillances.

Control room shift turnovers were orderly and briefings of individual operators were adequate. The NRC inspectors observed that the oncoming shift conducted another briefing for all operators after the off-going shift had left the control room. During these briefings (generally less than 5 minutes in duration), the operators discussed scheduled surveillance testing and general plant status information. During the QVFI, the plant experienced a loss of automatic feedwater control that led to a system transient. The operators quickly assessed the plant condition and responded to avoid a reactor trip on a low steam generator water level. The NRC inspectors observed that during the shift turnover briefing after the transient, incoming shift operators were very attentive to the briefing information. The licensee's Quality Assurance (QA) organization has audited this area several times and has not identified any problems with the adequacy and effectiveness of shift turnovers.

The NRC inspectors noted on several occasions that the operations manager and plant manager were in the control room observing shift turnover activities, other plant evolutions, and the shift supervisor's activities during different evolutions. At most times, an additional senior reactor operator (SRO) was available during the day shift. There also was good administrative-clerical support for the supervisors and operating staff. These support personnel appeared to remove some of the administrative burdens from the control room staff.

Management appeared to support quality operations and responded well to operators' recommendations concerning the use of operator aids in the control room. For example, a suggested operator aid, which consisted of a magnetized plastic card inscribed with the technical specification requirement and limiting condition of operation (LCO), was used on the engineered safety features actuation system bypass panel. This magnetized card covered the bypass key lock. When a system channel had to be bypassed, the operator aid had to be removed before inserting the key. The card then was placed in front of the control room operator to serve as a constant reminder of the condition that had to be monitored.

Management also has provided opportunity for operators to participate in a college training program. These personnel are sent to a local university to gain college credit towards meeting qualification requirements for a shift technical advisor (STA) position. There were times when several SROs on the same shift had the qualifications of an STA, which provided extra [illegible] of technical expertise to evaluate specific plant problems. The NRC inspectors observed that this program appeared to create higher morale and lower personnel turnover.

The NRC inspectors observed plant operator surveillance activities of technical specification requirements for safety-related systems and components. The inspectors also observed the licensee's QA overview of these operator surveillance activities. The QA personnel who were observed provided effective identification of problem areas during their overview. The NRC inspectors observed portions of 22 selected surveillance tests and all aspects of several other tests. Qualified personnel performed the tests and properly calibrated required test instrumentation, and the resulting data met the requirements of the Technical Specifications. When discrepancies were identified, they were rectified and the systems were properly returned to service. QA personnel were present while NRC inspectors observed surveillance tests.

The NRC inspectors watched tag out and equipment restoration on several occasions. The tag out and restoration processes, including briefings, were well understood by all operators who were involved. The NRC inspectors observed that QA personnel regularly reported and followed up on findings in these areas.

The NRC inspectors also watched operators perform a cycling test of the main turbine valves in accordance with Procedure STS AC-001, Revision 5. This surveillance test demonstrates the operability of the turbine overspeed protection system as required by Technical Specifications. The operators competently performed the test, and they adhered closely to the procedure. The test was satisfactorily completed without any irregularities or component malfunctions.

The NRC inspectors performed an auxiliary feedwater (AFW) system walkdown with a reactor operator using Procedure CKL AL-120, Revision 10, "Auxiliary Feedwater Normal Lineup," and piping and instrumentation diagram (P&ID) drawing M12AL01(Q), Revision 0. The NRC inspectors determined that the actual system configuration agreed with the P&ID drawing and that the operator appeared knowledgeable of valve locations and proper valve positions. The valves were found to be free of corrosion, locked if required, and positioned in agreement with the P&ID and the procedure. During the walkdown, the NRC inspectors identified four valves (EF-V077, FC-V115, AB-V085, GF-V009) with no labels and one valve (AL-V035) that contained a small packing leak. The reactor operator noted all deficiencies and they were corrected after being discussed with plant management.

During the AFW walkdown, the NRC inspectors noted that the position of the turbine-driven AFW pump (TDAFWP) speed set point on the auxiliary shutdown panel did not agree with required TDAFWP set point noted in Procedure CKL AL-120. The required speed set point was 3850 rpm, while the actual control set point was 5750 rpm. The inspectors discussed this discrepancy with cognizant instrumentation and control (I&C) personnel, who explained that the TDAFWP shutdown panel controller output signal to the pump is 3850 rpm, regardless of the higher set point. I&C personnel also provided, as a verification of this condition, the results of the testing of the TDAFWP controller conducted on October 23, 1987, in accordance with Procedure INC L-1000, Revision 2. The NRC inspectors discussed the TDAFWP speed set point with several operations personnel and determined that operator knowledge of equipment operation was acceptable.

2.1.2 Results Summary

The NRC inspectors observed during multiple dayshifts and backshifts that control room operators conducted themselves in a professional manner. Operators appeared to be attentive, were knowledgeable of plant status, and performed testing correctly with close adherence to procedures. The NRC inspectors verified that QA personnel did observe performance of several operational and maintenance work activities. The NRC inspectors' observations relating to the shift turnover briefings emphasize the need for operator attentiveness at all times.

2.2 Quality Assurance and Control Activities

2.2.1 Inspection Results

The NRC inspectors observed QA activities, interviewed QA personnel, and reviewed applicable QA audit and surveillance reports in the operations area. Specifically, the inspectors observed quality control (QC) (a part of Quality Department at Wolf Creek) involvement in maintenance and testing of reactor trip breakers and QA personnel performing a followup surveillance. The NRC inspectors also reviewed audits and surveillance reports that involved operations activities, as well as those covering general work control.

QA personnel audited normal, backshift, and weekend activities and surveyed operations activities. The NRC inspectors observed that QA personnel were knowledgeable and competent in their audits and surveillances and maintained an adequate mixture of direct QA observation of operational activities and review of documentation.

The NRC inspectors reviewed audit reports that spanned approximately 2 years. The quality of the reports and types of observations had recently improved, covering more of the actual performance of the activity rather than verifying strict compliance to procedures. An essential elements book was written and implemented by the QA department to ensure that the essential elements of test procedures were critically analyzed by a QA auditor during his or her verification activities.

The NRC inspectors observed that during the performance of maintenance on the reactor trip breakers (Work Request 50762-88), QC personnel were present and verified the completion of several in-process inspection hold points. In addition, the NRC inspectors accompanied a QA inspector during a followup surveillance of plant equipment, both safety related and nonsafety related, and of general plant conditions. This surveillance was performed to verify that corrective actions for deficiencies identified in QA audit TE53359 S-1627, "Control of Plant Equipment," had been implemented.

During the followup surveillance, the licensee's QA inspector identified several unacceptable conditions, including one that involved the storage of safety-related snubbers in the auxiliary building. More specifically, approximately nine mechanical snubbers had been functionally tested in early May 1988 and three had failed. Although all nine of the snubbers were appropriately tagged, the licensee did not segregate the failed snubbers from the snubbers that passed testing. In addition, all nine snubbers were stacked together in an area not designated for storage. When the NRC inspector questioned the acceptability and adequacy of this condition, licensee management had the snubbers moved to a proper storage area used for safety-related equipment. The licensee's QA inspector also identified a leaking valve on the second stage feedwater reheater drain tank. This valve, AFV 944, is a level switch isolation valve, and it contained a body-to-bonnet steam leak. The QA inspector reported this condition and subsequently the valve was repaired with furmanite to stop the leak. It was apparent to the NRC inspectors that the QA inspector was knowledgeable of proper plant conditions and of the need to promptly report results to management.

The NRC inspectors also reviewed approximately 10 recent QA audit and surveillance reports. One of these reports, QA Audit TE50140 K-192, "Corrective

Actions," identified O-rings in the solenoid operators of post-accident sampling system containment isolation valves that were not environmentally qualified (EQ). Licensee personnel discovered the O-rings that were not EQ in November 1987 during the implementation of a plant modification request (PMR 1844). This modification involved changing valve solenoid springs in several valves. During the implementation of PMR 1844, a maintenance crew mistakenly disassembled the solenoid operator of a valve (GS-HV-013) not requiring modification. The crew realized their error, and they also identified that the solenoid operator contained EPR-type O-rings that were not EQ for that specific application. Corrective work requests were written to inspect the solenoid operators and to replace the EPR O-rings that were not EQ with EQ grafoil O-rings, as necessary. The following valves were inspected: containment hydrogen control valves GS-HV-4, GS-HV-5, GS-HV-9, GS-HV-13, GS-HV-14, and GS-HV-18; nuclear sampling valves SJ-HV-3, SJ-HV-4, SJ-HV-5, and SJ-HV-128; and steam generator blowdown valves BM-HV-35, BM-HV-36, BM-HV-37, and BM-HV-38.

The QA organization issued a defect/deficiency report (D/DR 87-132) after discovering that the O-rings in the valves were not EQ. The QA organization also issued a quality plant deviation (QPD) and a programmatic deficiency report (PDR OP87-111) to address the disassembly of the wrong solenoid operator during implementation of PMR 1844. An engineering evaluation (87-SJ-10) was performed to determine the effect of having the O-rings that were not EQ in the valves. The results of the engineering evaluation showed that moisture or water that might intrude into the solenoid operator if an O-ring that was not EQ failed would not affect the valve's pressure retaining function; however, moisture could cause the valve to remain in the failed-closed position upon receipt of a containment isolation signal and not allow the valve to reopen to operate the post-accident sampling system. All of the work requests for the affected valves had been completed at the time of the engineering evaluation. It could not be determined how many of the 14 valves in question had contained O-rings that were not EQ because the O-rings in all valves were changed and the licensee did not document which of the valves had the O-rings that were not EQ.

The NRC inspectors determined that the valves were originally delivered with EPR O-rings that were not EQ, but were subsequently modified by Design Change Package (DCP) CS-90-W, Field Change Work Request (FCWR) FJ 603A-02, construction work permits CWP BM-212-E, CWP-GS-65I and work request WR698-85. At the time of the QVFI, it was unclear whether the O-rings had actually been replaced during implementation of the work permits and request or whether additional work on solenoid operator O-rings had been performed on the valves after the original issuance of the CWPs and WR. However, it is apparent that during the time when FJ 603A-02 was issued and CWP BM-212-E, CWP-GS-65I and WR 698-85 were all completed by January 21, 1985, the maintenance organization did not adequately accomplish the specified activities and the QC organization failed to verify, during their reviews and inspections, that the proper EQ O-rings had been installed. The licensee's actions to ensure that the deficient valve operators had EQ O-rings installed corrected the immediate problem. However, the NRC inspectors determined that the licensee had not investigated the underlying cause which permitted installation of O-rings that were not EQ to remain installed in the solenoid operators. This failure to determine the underlying cause of the condition is considered a potential enforcement finding (Item No. 88-200-1a). This issue is also addressed in NRC RIV Inspection Report 50-482/88-19 and will be followed up by Region IV.

2.2.2 Results Summary

The NRC inspectors determined that QA activities generally were conducted in a performance-oriented manner by qualified individuals.

2.3 Operations Training

2.3.1 Inspection Results

The NRC inspectors reviewed licensed, non-licensed, and craft training practices. The NRC inspectors' interviews with instructors indicated that the instructors were competent and professionally trained. Instructor performance is evaluated by the manager of training as well as by peer, self, technical peer, and supervisory personnel. Each instructor had been appropriately certified for the activities he or she was performing. There currently are four positions for licensed instructors, two were filled by qualified contract personnel, and two were vacant. The training staff and the instructional staff appeared to be dedicated, professionally competent, and responsive to student concerns and needs.

2.3.2 Results Summary

The NRC inspectors were concerned that two vacancies in positions for licensed instructors exists in the licensee's training department. This issue was discussed with licensee management to emphasize the importance of training and the need for a fully staffed training department.

3 PLANT MAINTENANCE

3.1 Maintenance Activities

3.1.1 Inspection Results

The NRC inspectors observed maintenance activities on a pressurizer code safety valve and a component cooling pump and evaluated the engineering support activities for maintenance on a pressurizer spray valve. The inspectors reviewed the following attributes of each maintenance activity: quality of instructions and worker training, familiarity of worker with the task and with tools and equipment, listing of task precautions, adherence to procedures, and QC involvement in the activity.

3.1.1.1 Pressurizer Spray Valve

The NRC inspectors reviewed engineering calculations generated by Nuclear Plant Engineering personnel in support of the encapsulation of a pressurizer spray valve packing box. The encapsulation was necessary to control a reactor coolant leak from the packing box assembly. The NRC inspectors determined that the calculations were detailed and accurate. Engineering personnel performed a thorough analysis that demonstrated good support of this maintenance activity.

3.1.1.2 Pressurizer Code Safety Valve

The NRC inspectors watched the maintenance technician set up and clean the components on a pressurizer code safety valve in preparation for disassembly and rework. The work was well organized and managed. The NRC inspectors also reviewed the applicable maintenance procedure to be used for this activity and determined that the detail, references, precautions, tool requirements, and other important data were adequate.

3.1.1.3 Component Cooling Water Pump Disassembly

The disassembly of the component cooling water (CCW) pump was a relatively complex task that relied heavily on skill-of-the-craft. The work instruction consisted of six general steps on the work request form and a reference to an attached photocopy of a section of the pump vendor's manual. The procedure used for disassembly was also photocopied from the pump vendor's manual.

During the work to remove the pump's bearing, the NRC inspectors observed that the maintenance technician was using a bearing puller on the bearing while heating the bearing housing with a gas flame torch. The technician involved was knowledgeable of the process, but not of potential effects that heating might have on the material characteristics of the bearing and pump shaft. No method was specified, nor was a contact thermometer on hand to determine the temperature of the heated parts. The NRC inspectors noted that the instruction to remove the bearing simply stated "remove the bearing." The work instruction did not include a caution statement addressing the potential damage to the pump shaft or bearing, heating instructions, expected temperature for bearing release, or maximum temperature recommendations.

QC inspectors were not present during this activity because it was not considered a detailed step requiring a QC hold point. Apparently, the quality organization responsible for procedural reviews determined that this bearing removal did not require additional details and that skill-of-the-craft was adequate.

The NRC inspectors discussed the lack of temperature limits and a heating process description and control with the procedure writing group in Maintenance Engineering. In response, the engineering supervisor stopped maintenance activities until the pump vendor could be consulted. Following consultation with the vendor, a bearing surface temperature limit of 750°F was specified, an expected bearing release temperature of 300° to 500°F was established, and heating process instructions were provided in a revision to the work request. The revision also indicated methods for monitoring bearing and shaft temperatures.

This CCW pump bearing removal activity indicated a weakness in the work process with regard to the appropriatness and adequacy of procedures and work instructions and is considered a potential enforcement finding (Item No. 88-200-2). Additionally, the assumption that skill-of-the-craft was sufficient for this activity was not prudent and demonstrated poor communication between procedure writers and task performers (Observation Item No. 88-200-3).

3.1.2 Results Summary

The NRC inspectors concluded that general maintenance technician performance was good, QC presence during performance was adequate, maintenance craft knowledge and experience levels were adequate, but work instructions, especially with regard to limitations and precautions, were weak.

The NRC inspectors determined that the probable causes of the work instruction weaknesses were the informality of work instructions, unfamiliarity of procedure writers with the task to be performed, inadequate attention to detail, and a lack of feedback from maintenance personnel to cognizant engineers on problems they encounter and recommendations to improve the instructions.

3.2 Control Building Heating, Ventilating, and Air Conditioning (HVAC) System

3.2.1 Inspection Results

From 1985 until now, the control room ventilation isolation signal (CRVIS) system has been activated 72 times as a result of spurious signals from the chlorine monitor system, and radiation detectors and other components in the HVAC system. More specifically, 28 of the CRVIS actuations have been attributed to malfunctions of the chlorine monitor system and the remaining 44 to problems with the radiation detection system, electrical circuit breakers and dampers within the HVAC system. The following sections detail the inspector's review of the three apparent contributors to the CRVIS actuations.

3.2.1.1 HVAC Breakers

The NRC inspectors reviewed records pertaining to problems with the control building HVAC circuit breakers. In early 1985, the licensee's Maintenance Engineering organization identified nuisance tripping of the HE3-B100-0501 ITE breakers at their respective motor control centers (MCC). An engineering evaluation request (EER 85-GK-08) was prepared by Maintenance Engineering on July 26, 1985. The resulting engineering evaluation, completed on November 27, 1985, stated that new breakers would be ordered with a specified instantaneous trip setting.

The NRC inspectors determined that the licensee had received the breakers ordered by engineering, but had never installed them in the designated system. Since the maintenance organization was not notified that the breakers had been received, their work request records indicated that this item was open because the parts were not available. The NRC inspectors determined that of the two breakers ordered for this system, one was in the warehouse and the other had been used in another system and not installed into the appropriate MCC as specified in engineering disposition REDA 0-E-1324-GK. It appeared that no one was tracking this item to ensure that the replacement breakers were installed as directed by engineering. This failure to take the specified corrective actions regarding the HVAC electrical system breakers malfunctions is considered a potential enforcement finding (Item No. 88-200-1b). In response to this issue, the licensee has committed to evaluate the existing engineering evaluation request tracking system.

3.2.1.2 HVAC Dampers

The inspectors reviewed records pertaining to problems with the control building HVAC dampers. The records indicated that during routine work, maintenance engineering personnel found that the HVAC dampers were not aligned as required by the design drawing. Although Maintenance Engineering determined that the observed misalignment was the cause of the damper failures, it was not evident whether Maintenance Engineering considered the cause of the misalignment during the investigation of the damper problem. Additionally, the investigation into the cause of the failures did not consider whether the multiple CRVIS actuations also were contributing to the damper problem. These failures to fully investigate the underlying causes of the multiple HVAC damper failures is considered a potential enforcement finding (Item No. 88-200-1c).

3.2.1.3 Control Room Habitability System Chlorine Monitors

The NRC inspectors reviewed the specification for the replacement chlorine detector monitors that are part of the control room habitability system and verified that the site-specific technical requirements for the monitors were defined within the specification criteria. The NRC inspectors also reviewed the engineering design calculations to ensure that the technical specification requirements were considered when evaluating the new design criteria.

The chlorine monitors are essential elements of the control room habitability systems. These habitability systems permit access to and occupancy of the control room during normal plant operations as well as during and following emergency conditions. They also are designed to enable the plant operators to achieve and maintain the plant in a safe shutdown condition following a design-basis accident (DBA)

As discussed previously, the chlorine monitors have caused 28 actuations of the control room ventilation isolation signal system (CRVIS) since 1985. Eighteen of these actuations were due to paper tape problems, seven were due to signal spikes from the chlorine monitors, and three were attributed to causes such as manual actuation and personnel errors. Operations personnel currently are required to survey the chlorine monitors twice per shift to look for indications of a possible malfunction.

The NRC inspectors reviewed a recent engineering study that had been conducted to provide solutions to prevent further malfunction of the control room chlorine monitoring system. This study indicated that the major problems with the chlorine monitors were tape failures, electrical failures, spurious spikes with tape failures, and failures of lamps. The licensee recently issued a work order to remove a WISA pump from its present location in the chlorine monitor unit to a remote location because the licensee believed that WISA pump vibrations may have been contributing to the problems. The results of this modification will not be known until sufficient operating time has elapsed.

The licensee also plans to replace the 7040 MDA model monitor with a commercial grade Delta chlorine detector system during the next outage scheduled for the last quarter of 1988. The Delta system is to be dedicated and qualified during the third quarter of 1988. In addition, the licensee has ordered a Sensidyne chlorine detector system to back up the Delta chlorine detector system. The NRC inspectors determined that the licensee's activities to replace the present MDA chlorine monitor system with Delta and Sensidyne systems were positive actions to resolve the problem.

The licensee has experienced a large number of CRVIS actuations resulting from the chlorine monitoring system malfunctions without aggressively pursuing resolution of the problem until recently. Because of the large number of CRVIS actuations attributed to chlorine monitor malfunctions since 1985 and the apparent slowness with which the licensee has taken action to correct the problem, this matter is considered a potential enforcement finding (Item No. 88-200-1d).

3.2.2 Results Summary

On the basis of the above, the licensee's program for determining the underlying causes of plant system and component failures and malfunctions needs strengthening. The fragmentation of responsibility for implementing the WCGS corrective action program appears to be contributing to the program's weakness. With the exception of maintenance technicians, no single organization has been given the responsibility to technically analyze failures and malfunctions to determine their underlying causes. Additionally, the licensee's investigations of component failures and malfunctions do not always consider their effect on interrelated systems.

Without adequate cause evaluation information, thorough analysis of failures and malfunctions cannot be made and the trending programs become merely failure frequency indicators. Trending information should be used to increase the reliability of plant systems through early detection of repetitive component failures.

3.3 Maintenance Measuring and Test Equipment

3.3.1 Inspection Results

The NRC inspectors reviewed the QA activities associated with measuring and test equipment (M&TE). The inspectors selected a review sample of M&TE used on various maintenance activities to determine the adequacy of the out-of-tolerance evaluations, of the historical documentation of M&TE use (use-history), and of the QA corrective action process. Additionally, the inspectors reviewed 10 randomly-selected out-of-tolerance evaluations to verify timeliness and technical adequacy.

The NRC inspectors determined that, with one exception, out-of-tolerance evaluations were performed in a timely manner and were technically adequate. That exception, an evaluation for micrometer No. WC-6710, indicated that past usage of the lost instrument was acceptable because the previous two annual calibrations were within acceptable tolerances. In this case, better assurance of the micrometer's accuracy during previous use would have been provided by a remeasurement of affected activities to verify if the previously taken meaurements were within expected ranges. The inspectors considered the micrometer example to be isolated.

3.3.2. Results Summary

The NRC inspectors determined that the measuring and test equipment program adequately supports ongoing maintenance activities.

3.4 Fire Protection System

3.4.1 Inspection Results

The licensee has experienced a high instance of alarms activating as a result of the malfunction of a specific type of microswitch used in the fire protection system. These microswitches are installed on various outdoor valves that are located above and below ground. The Maintenance Engineering organization issued EER 87-FR-06 on May 8, 1987, which stated that the present microswitches, Type PV IS-B, are routinely found corroded and are being used in applications for which they were not designed. The NRC inspectors reviewed 16 recent work requests associated with microswitch failures and found that the microswitches continue to be misapplied. At the time of this inspection, the licensee had not taken action to stop using the microswitches in applications for which they were not designed. This failure to take actions to resolve the apparent misapplicaton of the microswitches is considered a potential enforcement finding (Item No. 88-200-1e).

3.4.2 Results Summary

The NRC inspectors determined that the fire protection system was adequate. However, the inspectors were concerned that the control room alarms that resulted from the malfunctioning microswitches may desensitize the operators to an actual fire protection system actuation.

3.5 Emergency Diesel Generator Vendor Service Information Letters

3.5.1 Inspection Results

The NRC inspectors reviewed several service information letters (SILs) to determine whether proper evaluation and implementation of any necessary component inspections and modifications had been performed by the licensee. These SILs were issued by the emergency diesel generator (EDG) vendor, Colt Industries, to convey vital service information to its customers.

During an interview, licensee personnel told the NRC inspectors that Colt SILs are considered vendor technical information, which is to be reviewed and evaluated under Wolf Creek's Industry Technical Information Program (ITIP). The ITIP was established in response to NRC Generic Letter 83-28, Section 2.2.2. However, when the NRC inspectors asked to review the evaluation of Colt SILs conducted under the ITIP, licensee personnel gave the NRC inspectors an interoffice memorandum (No. AD 87-0373) dated November 9, 1987, which stated that no Colt SILs had been transmitted to ITIP personnel for their review and evaluation because of miscommunications between the vendor (Colt), the plant's architect-engineer (Bechtel), and licensee personnel. The memorandum also requested that Colt be contacted to determine which SILs were applicable to Wolf Creek and to send them for immediate review by ITIP personnel.

Colt determined that there were five SILs that pertained to the EDGs supplied to the licensee. Licensee personnel stated that they had received the SILs from Colt in January 1988. However, at the time of this inspection, the NRC inspectors found no formal review or evaluation of the five SILs had been performed by the licensee. Further, the inspectors determined that the licensee had not received three other SILs that pertained to the Wolf Creek EDGs. This failure to obtain all relevant Colt SILs, review them to determine their applicability to WCGS, and evaluate their relevance to the Wolf Creek EDGs is considered to be a potential enforcement finding (Item No. 88-200-4). Subsequent to this finding, the licensee issued Programmatic Deficiency Report OP-88-124 and issued an engineering evaluation request to determine if additional Colt SILs, which were applicable to Wolf Creek, existed and had not been received by ITIP personnel.

The NRC inspectors reviewed Colt SIL, Issue 7 (December 16, 1985), entitled "Intercooler Spacer Bar," to determine if information therein pertained to the Wolf Creek EDGs. This SIL, which had not been received by the licensee, addressed a potential problem associated with the spacer bar supporting the side-mounted turbocharger intercoolers and noted that the spacer bar mounting bolts should be periodically checked for tightness.

The NRC inspectors performed a field walkdown of the A and B EDG intercooler supports and associated cooling water piping. During the walkdown, the inspectors noted that all four turbocharger cooling water piping lines were missing a seismic and vibration control pipe support that was required to be installed by vendor's design drawings. In response to this observation, the licensee contacted the vendor (Colt Industries) to determine if the turbocharger cooling pipe could perform its intended function without the seismic and vibration control supports and whether the turbocharger cooling pipe would experience cracking or the flange bolts would loosen as a result of excessive vibration. The vendor referred the licensee to Colt Industries' Engineering Report No. M-018-0367-02, "Seismic Calculations for Skid Mounted Piping." A table in this report indicated that the support bracket would be required for the turbocharger cooling piping in a seismic event if the length of the piping was greater than 60.7 inches. The licensee measured the subject piping and found that it was 56 inches in length; thus concluding that the turbocharger cooling pipe could perform its intended function during a seismic event without the support bracket. The licensee gave the NRC inspectors a draft copy of their engineering seismic calculation, which also indicated that the pipe did not require the support to withstand seismic loading.

Even though the available engineering data did not support installation of the supports for seismic reasons, Colt urged the licensee to install the four missing supports to ensure that vibrations from the operating diesel engine would not cause degradation of engine components. In addition, Colt recommended that the licensee visually inspect the pipes for cracking and a loss of jacket cooling water and perform a torque inspection for all associated pipe flange bolts.

The licensee took immediate actions to fabricate and install the four pipe supports and performed the inspections recommended by Colt. During those inspections, quality control inspectors found that the turbocharger cooling piping on the A EDG contained a weld defect. This item was referred to engineering for further evaluation. In addition, when it was determined the two of the flange bolts were torqued below minimum requirements, a work request was issued to retorque all of the affected pipe flange bolts for both emergency diesel engines. Before the conclusion of the inspection, the licensee further committed to perform nondestructive examinations on all four turbocharger cooling water pipes and a vibration test and analysis to determine if there were any additional adverse effects on the cooling pipe caused by operating the EDGs without the supports.

At the time the EDGs were originally constructed at Wolf Creek, the turbocharger cooling water piping vibration supports were not installed as required by the vendor's design drawing. During the installation work, licensee personnel who were responsible for verifying that the EDGs were properly constructed did not ensure that the supports had been installed. This failure to verify that the as-built configuration of the EDGs was consistent with the Colt design drawing is a potential enforcement finding (Item No. 88-200-5).

3.5.2 Results Summary

The NRC inspectors identified several instances where the licensee did not obtain and evaluate all the applicable EDG vendor information (SILs). In part, this contributed to the four missing pipe supports for the cooling water piping lines not being discovered by the licensee. Although the supports were not necessary for seismic support, the vendor did recommend that they be added for vibration reasons. QC inspections of the EDGs during this inspection did reveal that excessive EDG operational vibrations had caused the pipe flange bolts to loosen to the point where they did not meet torque requirements. These issues point to the need for additional attention to detail in the area of vendor interface.

3.6. Diesel Generator Jacket Water Pressure Transmitters

3.6.1 Inspection Results

The NRC inspectors performed a walkdown of the A and B EDGs and their associated support systems. During the walkdown, the NRC inspectors noticed plant modification request (PMR) Tag No. 20315, dated April 11, 1986, adjacent to the jacket water pressure indicator gauge on the local control panel for the A EDG.

The information on the tag indicated that a pulse in the gauge's sensing line was causing a false indication on the pressure gauge. The NRC inspectors went to the local control panel for the B EDG to determine if the same condition existed. They saw two information tags located next to pressure gauges for the jacket cooling water and the jacket water intercooler. Both information tags indicated that there was a sensing line pulsation problem and that the lines were valved out to isolate the system and stop spurious alarms in the control room during system testing. The NRC inspectors asked a plant operator if it was possible that the sensing line for the indicator gauge on the A EDG was isolated even though there was no indication of such on the PMR tag. The operator stated that the PMR tag did not serve that purpose and that the line for the A EDG should not be isolated. However, when the NRC inspectors and the operator examined the line, they found it had been valved out and isolated. In response, the operator notified the SRO on duty and replaced the PMR tag with one containing the correct line configuration information. Because the licensee took immediate action to correct the problem and because the line was used for indication of system operating parameters, the inspectors have considered this issue adequately resolved.

The NRC inspectors interviewed cognizant instrumentation and control (I&C) personnel to determine why the false indication conditions existed and what had been done to correct the problem. Previously, a temporary modification was implemented to install pressure damping devices in the sensing line. The dampers alleviated the problem until they became clogged with impurities from the jacket cooling water. Subsequently, the dampers were removed. The licensee then initiated EER 87-KJ-01 (June 9, 1987) to resolve the condition. The EER contained information indicating that the problems with the pressure transmitters resulted from pressure pulsations in the jacket water sensing line side of the transmitter. In the disposition of the EER, Plant Engineering recommended remounting the pressure transmitters and placing dampers or similar flow restriction devices adjacent to the transmitter where the transmitter sensing line ties into the pressure portion of the system. This modification, when implemented, will shorten the length of line between the transmitter and damper and reduce the amount of impurities that could clog the dampers. I&C personnel stated that, although this modification is planned, the problem is still ongoing. The licensee has not to date considered the cause of the pulsations and the effect of the proposed corrective actions. This failure to aggressively pursue the cause and take action to stop the sensing line pulsations that have existed since 1986 is considered a potential enforcement finding (Item No. 88-200-1f).

3.6.2 Results Summary

The NRC inspectors determined that the licensee had not adequately addressed the malfunctions in the jacket water pressure sensing line and instruments of the EDGs. Since initial discovery of the problem in April 1986 to the time of the QVFI, the licensee has not aggressively pursued the cause of the pulsations in the system nor have they implemented timely, effective corrective actions to ensure accurate and reliable system performance. Disregard of this instrument's inability to perform its intended function is not an attribute of prudent, safe operation of the EDG system.

4 INDEPENDENT SAFETY REVIEW ORGANIZATIONS

The NRC inspectors reviewed the activities of Wolf Creek's independent safety review groups to determine their effectiveness and contribution to the plant's safe and reliable operation.

4.1 Plant Safety Review Committee (PSRC)

4.1.1 Inspection Results

The NRC inspectors reviewed the minutes of six PSRC meetings (306, 316, 317, 319, 320, and 322), interviewed selected personnel with regard to the PSRC activities, and attended a PSRC meeting (No. 322) on June 14, 1988.

The PSRC function is specified by Procedure ADM 01-002, Revision 16, "Plant Safety Review Committee." The procedure implements the requirements of Technical Specification 6.5.1, "Plant Safety Review Committee (PSRC)." The PSRC meetings were conducted routinely at weekly intervals, which is more frequently than required by the Technical Specifications. Additional meetings were scheduled when deemed appropriate. The QA manager, or a designated alternate, normally attends the scheduled PSRC meetings, even though the QA manager is not a member.

The NRC inspectors determined that all but two of the selected PSRC members had the experience and equivalent training normally required to take an examination for a senior reactor operator's license at Wolf Creek. The two PSRC members with less extensive training were the Manager of Maintenance and Modifications and the Manager of Plant Support. The inspectors discussed upgrading the training of these two managers with the licensee.

The NRC inspectors reviewed the materials discussed during the PSRC meeting (322) conducted on June 14, 1988. During the meeting, plant modification request PMR 02577, Revision 0, "Penetration Boundary Change," was reviewed. The PMR had been processed in response to corrective work request (WR 00688-88) dated February 9, 1988. The WR was written to document that the top 6 inches of Radflex material was missing from penetration OP 142S1099 located on elevation 2026' of the auxiliary building. The shift supervisor declared the penetration operable on February 9, 1988. The initial review of the degraded condition of the sealant was completed on February 18, 1988, and resulted in a "use-as-is" disposition of the WR. The basis for the use-as-is disposition was that there was enough Radflex material remaining to allow sufficient fire rating but not enough for a radiation barrier. As a result, the design of the penetration seal was revised from an RB-9 type (Radflex) to an M-9 (fire seal).

The followup engineering disposition regarding the condition of the penetration was completed on May 3, 1988, and concluded that the floor at elevation 2026' separates fire area boundaries and requires a 3-hour fire-rated penetration seal. Because of the uncertainty of the current consistency of the Radflex material in penetration OP 142S1099, engineering could not establish that the penetration would meet these fire qualification testing requirements.

WCGS's Updated Safety Analysis Report (USAR), Section 9.5, Table 9.5.1-3, requires that all fire barriers and their penetrations separating safety-related areas from those that are not safety related or separating portions of redundant systems important to safe shutdown shall be operable at all times. Should one or more be found to be inoperable, a continuous fire watch on one side of the affected barrier or an hourly fire watch patrol must be established within 1 hour. The inspectors discussed the May 3 engineering evaluation and the degraded condition of the fire seal with the licensee. On June 14, 1988, the licensee issued Fire Protection Impairment Control Permit No.88-244 to establish a firewatch. In effect, a fire watch should have been established within 1 hour from the time the fire seal was determined not to meet fire qualification testing requirements. This failure to implement the required fire watch between May 3 and June 14, 1988, is considered a potential enforcement finding (Item No. 88-200-6).

4.1.2 Results Summary

The NRC inspectors determined that, with the exception that a required fire watch for an unqualified penetration fire barrier was not established, the PSRC function was established and functioning as required by Technical Specifications.

4.2 Nuclear Safety Review Committee (NSRC)

4.2.1 Inspection Results

The NRC inspectors reviewed the minutes of NSRC meetings conducted in 1987 and 1988 and interviewed selected personnel with regard to NSRC activities.

The NSRC function is specified by Policy No. II.13.0, Revision 3, "Nuclear Safety Review Committee Charter." The policy implements the requirements of Technical Specification 6.5.2. Document reviews and discussions revealed that the meetings were scheduled and conducted more frequently than required--generally three or four times per year. NSRC meetings are routinely conducted at the site training center and include a scheduled plant tour. Also, the members can independently review specific areas of plant operations, such as operations, chemistry, and health physics. The requirements of NSRC audits is addressed in detail, including overall responsibility, planning and implementation, audit reports, and resolution of findings. The audits and audit results are maintained in an action item list, as reflected in the NSRC meeting minutes.

4.2.2 Results Summary

The NRC inspectors determined that the NSRC consisted of technically capable personnel who fulfill the requirements of the Technical Specifications. The NSRC has provided upper management with technically sound recommendations concerning plant safety and reliability and are functioning as an effective quality verification organization.

4.3 Nuclear Safety Engineering (NSE)

4.3.1 Inspection Results

The NRC inspectors reviewed selected NSE reviews and evaluations to determine the effectiveness of NSE as an independent quality verification organization.

The NSE function is specified by Procedure KP-750, Revision 0, "Statement of Responsibilities Nuclear Safety Engineering." The procedure implements Item I.B.1.2 of NRC NUREG-0737, Technical Specification 6.2.3, USAR Chapter 18.1.7.2, and outlines actions in response to NRC Generic Letter 83-028. NSE performs surveillances of plant activities in accordance with the requirements of Procedure KP-751, Revision 0, "Surveillance of WCGS Activities by Nuclear Safety Engineering." The procedure provided definition, responsibilities, and the scope of the surveillance activities for NSE.

The NSE also reviews almost all operational information concerning other commercial nuclear power facilities. It routinely receives all reactor trip data and is required to complete the independent review of all unscheduled reactor trips before reactor restart if the trip was complicated by other plant perturbations.

Recently, the NSRC requested NSE to investigate a 4-percent indicated decrease in total reactor coolant system (RCS) flow. NSE determined that an analysis of the calibration data for the RCS narrow-range resistance temperature detectors (RTDs), which were used to establish core enthalpy rise, was required because an increase of 1.5 to 2.0°F had been identified. The review of the RTD calibration data taken during the 1987 outage was compared to the data taken during initial startup in 1985. The comparison indicated (1) a much wider variation between the hot leg RTDs (but not exhibited between the cold leg RTDs) and (2) a disparity between the hot leg and cold leg RTDs. The wider variation exhibited by the hot leg RTDs and the disparity between the hot and cold leg RTDs indicated that the hot leg RTDs output signals had drifted differently than the cold leg RTDs, possibly as a result of the steep temperature gradients (and resultant thermal stress) experienced by the hot leg RTDs following a reactor trip (a large number of which occurred during the first and second year of plant operation).

The licensee had implemented a number of actions to attempt to reduce the RTD errors, including the Westinghouse error analysis methodology. These actions resulted in a reduction in the RTD errors and an increase in the indicated (calculated) RCS flow. However, the NRC inspectors were concerned that the routine use of the Westinghouse error analysis methodology (cross-calibration of RTDs and development of correction factors) and the utilization of RTD vendor supplied resistance (R) versus temperature (T) curves may not be conservative, in that the RTDs at Wolf Creek (or any other facility which uses such methodology) may never be calibrated to a known standard to ensure generic senser drift does not occur during the 40-year lifetime of the plant. Wolf Creek does not use the RTDs installed in thermowells to calibrate, under controlled conditions, the RTDs in the protection system (immersion-type RTDs). These system RTDs have not been checked to a known standard, directly or indirectly, since initial installation.

Items 7 and 8 ("Overtemperature Delta T" and "Overpower Delta T," respectively) in Technical Specification 3/4.3.1, "Reactor Trip System Instrumentation," Table 4.3-1, specify that a channnel calibration is to be performed at least once every 18 months. Technical Specification 1.5, "Channel Calibration," specifies in part that a channel calibration shall be adjusted, as necessary, such that the channel responds within the required range and accuracy to known values of input and shall encompass the entire channel including the sensors. The methodology used to calibrate the RTDs does not include checking the accuracy of the RTDs to known values of input (temperature). Shifts in the RTD calibration curve may not be detected in a timely manner, which may result in out-of-tolerance RTD output values (Observation Item No. 88-200-7). The NRC inspectors discussed this matter with the licensee; it will require further NRC NRR staff review.

4.3.2 Results Summary

The NRC inspectors determined that the NSE appeared to be an effective, technically-oriented organization. The NSE has provided management with extensive and accurate assessments of plant issues, such as the RTD cross calibration issue and the problems with the control room chlorine monitors.

5 INDUSTRY TECHNICAL INFORMATION PROGRAM (ITIP)

5.1 Inspection Results

The ITIP function is specified by Procedure KGP-1311, Revision 1, "Industry Technical Information Program." The ITIP implements the licensee's response to items addressed in NRC NUREG-0737, Item I.C.5, "Procedures for Feedback of Operating Experience to Plant Staff."

The NRC inspectors reviewed evaluations of twelve ITIP items received by the licensee, as well as selected monthly status reports, a recent QA audit report, the most recent effectiveness review report, and Procedure KGP-1311. The inspectors held discussions with selected licensee personnel with regard to ITIP activities.

The NRC inspectors' review of the completed ITIP evaluations indicated that the timeliness of the reviews had improved dramatically over the past 3 months. The timeliness issue was previously identified in QA Audit Report TE:50140-K202, dated March 23, 1988. The report specifically identified the lack of timeliness of the initial evaluations, a significant backlog of items requiring reviews, and the need to complete programmatic changes expeditiously. The evaluation review times have recently decreased from months to days. Discussions revealed that che licensee was applying additional effort to decrease the ITIP backlog and other programmatic improvements have been completed.

5.2 Results Summary

The NRC inspectors determined that weaknesses noted by the licensee's QA organization regarding the timeliness of reviews have recently improved. However, the importance of evaluating industry information on plant equipment and components in a timely way is necessary for reliable and safe operations. Section 3.5 of this report provides details of the ramifications when the ITIP fails to fulfill its required function. Other ITIP functions were implemented in accordance with applicable WCGS procedures.

6 ACTIVITY/EVENT REVIEW

The NRC inspectors reviewed the effectiveness of the licensee's quality verification organizations through the corrective actions associated with four specific activities: (1) emergency service water pipe wall thinning, (2) pressurizer spray valve replacement packing box, (3) reactor vessel head O-ring leakage, and (4) containment cooler A repair.

6.1 Emergency Service Water (ESW) Pipe Wall Thinning

6.1.1 Inspection Results

The NRC inspectors reviewed documents and interviewed licensee personnel with regard to pipe wall thinning experienced in portions of the ESW system in 1985 during normal system operations. Pipe wall thinning appeared to be caused by erosion/corrosion from combinations of elevated flow rates through throttled butterfly valves and the configuration of the ESW system.

With the exception of several short outages resulting from equipment malfunctions, the unit operated continuously until the commencement of the refueling outage in September 1987. The NRC inspectors reviewed a number of specific activities related to the corrective actions associated with pipe wall thinning. Work Request (WR) 00653-87 was issued on February 13, 1987, documenting the fact that the ESW piping below valve EFV-058 (throttled butterfly valve) was less than the specified minimum pipe wall thickness of 0.328 inches in numerous locations. The WR noted that the system was operable and the condition not reportable per 10 CFR 50.72. The WR was forwarded to Nuclear Plant Engineering (NPE) for evaluation and an engineering disposition was provided on February 19, 1987, specifying that repair of the minimum wall for pipe spool piece 1-EFOS-S-005/142 should be repaired per instructions in Plant Modification Request (PMR) 1903. PMR 1903 had been used to repair train "B" of the ESW system during the 1986 refueling outage. The weld overlay repair of the ESW piping was subsequently performed during June 26 to July 1, 1987. The required system leak test was performed on July 1, 1987.

In May 1987, the Nuclear Safety Engineering (NSE) group performed a surveillance (SSR 87-045) of selected activities associated with the ESW pipe minimum wall thickness deficiency. A draft report of SSR 87-045 was provided to the plant manager on May 14, 1987. In the draft, NSE noted that no justification for continued operation had been provided regarding the thin-wall ESW system piping in that the engineering evaluation request (EER) only addressed the final weld overlay repair condition. The NSRC chairman, made aware of the issue by the NSE, also pursued the questionable condition of train A of the ESW system. Independent calculations were also performed by the licensee's engineering staff that confirmed that the ESW did not meet all its design requirements. Wolf Creek Updated Safety Analysis Report (USAR) Section 9.2.1.2.1.1 states that the ESW piping and valves are designed to the requirements of ASME Section III, Class 3. Section 9.2.1.2.1.1 of the USAR states that the ESW is safety-related, is required to function following a Design Basis Earthquake (DBA), and is required to achieve and maintain the plant in a safe shutdown condition.

The report of SSR 87-045, dated June 4, 1987, identified three concerns regarding the handling of minimum wall work requests, including (1) the operability determination made by the shift supervisor, (2) availability of information to operations, and (3) a defined program for handling pipe erosion. Their report also stated that the current safety evaluation covers only the permanent repair and not the justification for the continued operability of the component during the interim period.

The NSRC, aware of the EWS wall thinning issue in May 1987 as a result of NSE involvement, held discussions with plant management, including the Chief Executive Officer (CEO), on June 2, 1987. The NSRC Chairman was intimately involved in the oral and written communications regarding the concerns raised by the NSE surveillance (SSR 87-045, dated June 4, 1987) performed in mid-May 1987. On June 18, 1987, the NSRC chairman established a special review group (two consultants) to evaluate the ESW wall thinning matter. The issue was discussed in detail in NSRC meeting 87-02, conducted on June 24 and 25, 1987, and specific recommendations were provided to the licensee's CEO by letter on July 10, 1987 for consideration. The recommendations included the following:

(1) Further encourage and formalize the communications process between project personnel in the area of nonconformance report (NCR) disposition and implementation.

(2) Require a study and documentation of the lessons learned as a result of this occurrence with ESW pipe wall thinning from February 1987 until the repair is completed. Additionally, the CEO was provided a copy of the NSE surveillance report of SSR 87-045, which could not be closed because corrective actions have not been completed.

The licensee's CEO provided a draft letter on the subject to the three vice presidents (engineering, quality, and nuclear operations) on July 28, 1987. The draft letter addressed three matters regarding the "determination of operability," including (1) the operations group responsibilities, (2) NPE evaluations of nonconforming conditions, and (3) provision of the NPE evaluation to operations for reassessment.

Subsequently, plant Administrative Procedure ADM 08-212 and Engineering Procedure KPN-314 were revised to provide an erosion/corrosion program and provide notification of operations by engineering where the system design function is adversely affected. Operations Order (OP) 87-110 was issued on July 29, 1987, to instruct that all thin-wall piping problems would be documented by an NCR in accordance with the procedure. To enhance communications between operations and engineering, engineering now attends work planning meetings. Licensee personnel stated that they will, in the future, request temporary relief from ASME Section XI requirements from the NRC, when required, and provide technical justification for continued operation for each relief request. NPE had responded to the NSE surveillance report (SSR 87-045) on February 2, 1988. Nuclear Operations had not responded at the time of this inspection.

The NRC inspectors determined that the licensee's actions were inappropriate immediately following the discovery of the degraded ESW piping on February 13, 1987. The engineering disposition completed on February 19, 1987, stated that the degraded ESW pipe downstream of valve EFV 058 required repairs because it did not meet ASME requirements and, therefore, may not be capable of performing its specified safety functions. At this time, the licensee did not declare the system inoperable and allowed it to remain inservice until repairs were begun on June 26, 1987. Wolf Creek Technical Specification Limiting Condition for Operation (LCO) 3.7.4 requires at least two independent ESW loops be operable. In addition, with only one ESW loop operable, the inoperable ESW loop must be restored to operable status within 72 hours or the reactor must be in at least hot standby within 6 hours and in cold shutdown within the following 30 hours. The licensee's failure to declare the ESW system inoperable and meet the requirements of LCO 3.7.4 is considered a potential enforcement finding (Item No. 88-200-8). This issue was previously discussed in Region IV Inspection Report 87-15, dated July 22, 1987.

6.1.2 Results Summary

The lack of timeliness and thoroughness of the operational response to the ESW pipe wall thinning and the apparent lack of coordination between the plant operations staff and the various technical support groups is considered a significant weakness in the licensee's operability determination process. This recurrance of a problem with ESW pipe wall thinning should have triggered an immediate response by the licensee to declare the system inoperable and procede with expeditious repairs to assure safe and reliable system operation.

6.2 Pressurizer Spray Valve Replacement Packing Box

6.2.1 Inspection Results

The NRC inspectors reviewed the documented work activities associated with the replacement of the pressurizer spray valve packing box. The work was completed on December 29, 1987, in accordance with work request WR 00101-87, dated January 1, 1987, and ASME Section XI Plan No. RR-87-074, dated September 21, 1987.

The NRC inspectors reviewed documents and interviewed licensee personnel with regard to the work completion review by the licensee for the pressurizer spray valve packing box replacement on March 18, 1988. The appropriate component qualification documentation, which was supposed to have been part of the procurement package from Westinghouse, had not been received. As a result, the licensee could not ensure that the assembly of the pressurizer spray valve packing box met ASME Section XI requirements. The missing documents were the required ASME Code Data Report, Certified Material Test Reports (CMTRs), and nondestructive examination (NDEP reports. Even so, the plant was restarted in late December 1987 with the ASME pressure boundary component of undetermined quality installed in the reactor coolant system. Engineering analyses did not determine that the the component was acceptable and ASME Section XI relief was not obtained until late May 1988 (WR 1285-88, dated March 17, 1988 completed on May 24, 1988; and PMR 02535, dated April 25, 1988).

The licensee initiated a Programmatic Deficiency Report (PDR MM 88-07, dated April 18, 1988) documenting a deficiency with the pressurizer spray valve packing box. The Manager of Purchasing and Material Services was given the assignment to coordinate and collect documentation of the completed actions in response to the PDR. The PDR addressed the WCGS program and Westinghouse procedure requirements and the immediate corrective actions to investigate the apparent Westinghouse error in omitting the appropriate component qualification documentation. In addition, a review of all ASME Code Section III documentation packages that had been received prior to installing an ASME Section III boundary item was to be performed and the spare spray valve packing box assembly procured at the same time as the installed component was to be rejected. The cause of the issue was to be determined and corrective actions to prevent its recurrence were to be performed. The NRC inspectors determined that all but two of the actions were completed. The rescheduled completion date for the open items was September 1, 1988.

The overall corrective actions included the review of other repair-replacement work packages to ensure that no other nonconforming safety-related equipment had been installed. This review revealed that 17 issues required comment resolution. However, none of these issues were deemed by the licensee to impact safety-related equipment operability.

The licensee plans to replace the nonconforming valve packing box assembly during the next outage of sufficient duration and proper plant conditions to permit replacement. The NRC inspectors concluded that the licensee's overall corrective actions for this issue appeared to be extensive and acceptable.

6.2.2 Results Summary

The licensee's failure to ensure that the spray valve packing box conformed to ASME requirements before plant startup has been addressed in NRC Inspection Report 50-482/88-15. The licensee's actions taken once the deficiency was identified were both timely and effective and should ensure that repetition will not occur.

6.3 Reactor Vessel Head O-Ring Leakage

6.3.1 Inspection Results

The NRC inspectors reviewed the documented activities associated with the reactor vessel head O-ring leakage event of December 26, 1987, through Janaury 21, 1988.

Leakage from the inner reactor vessel (RV) head O-ring occurred on December 26, 1987, during plant startup after the scheduled refueling outage. The inner O-ring leakoff path was isolated and the leakage system aligned to monitor the outer O-ring. Subsequently, additional leakage was detected when the leakoff temperature from the outer O-ring increased between January 19 and 21, 1988.

The reactor was cooled down and the RV head removed on January 26, 1988. Between January 26 and February 2, 1988, the RV head O-ring seating area was inspected and cleaned. The licensee determined that the RCS level was at too high a level during the previous RV head placement, allowing water to overflow into the O-ring channel and eventually leak into the leakage indication system. On February 2, 1988, the licensee commenced installation of the new O-rings with the RCS coolant level between 12 and 39 inches below the RV flange level. The RV head was set and the RV studs immediately installed and torqued while the RCS water level was held at 39 inches below the RV flange. Subsequent RV head O-ring performance has been satisfactory.

The NRC inspectors determined by document review and interviews that corrective actions associated with this event were extensive and included specific program and procedure improvements. Further, the installation of a new RV water level indication system during the next refueling outage was scheduled and should improve the control of the RCS water level during outages.

6.3.2 Results Summary

The licensee's inattention to detail during the installation of the RV head O-rings during the 1987 refueling outage resulted in a forced plant shutdown as a result of leakage of reactor coolant past the O-rings. The licensee's maintenance staff did not provide quality workmanship and the quality verification personnel involved did not identify the deficiencies, which would have prevented the forced shutdown. (Reference NRC Inspection Report 50-482/88-04) In addition, the removal and replacement of RV head O-rings resulted in many additional staff-rem exposure hours, which is a concern to the ALARA commitment (as low as reasonably achievable). The additional stress placed on plant personnel as well as systems and components as a result of the shutdown and followup activities also is significant from a safety standpoint.

The NRC inspectors determined that the overall corrective actions taken by the licensee regarding this matter appeared to be acceptable. These actions should provide improved cleanliness controls associated with the RV head, adequately inspected RV O-rings, and an enhanced RCS water level indication system. The RCS new water level indication system also will provide better control of the RCS water level.

6.4 Containment Cooler A Repair

6.4.1 Inspection Results

The NRC inspectors reviewed documented work activity associated with the repair of the A containment cooler to assess the work planning and the quality assurance/quality control (QA/QC) involvement and conducted selected interviews to provide an adequate understanding of the work plan and associated activities.

The A containment cooler leak was identified and a work request (WR 04105-87) initiated on October 16, 1987. The licensee conducted a substantial amount of planning, repair selections, engineering, repair work, and testing during October 20 to December 22, 1987. The repaired cooler was tested on December 1, 1987, the cooler support reinspected on December 14, 1987, and the cooler was returned to service on December 15, 1987. The insulation was replaced on December 18, 1987. The final maintenance review was performed by December 22, 1987, and the final quality review was complete on February 7, 1988.

Subsequent review by the licensee in April 1988 of the completed work package revealed two deficiencies: (1) no certified material test report from a qualified vendor was available for the brazing material used to repair the cooler coils and (2) repairs to the tubes of the heat exchanger coil were performed without a qualified brazing procedure and a procedure qualification record for the base metal and thickness required. The specific deficiencies were brought to the attention of engineering and the subsequent engineering dispositions concluded that the items were acceptable. Programmatic deficiency reports were written on May 26, 1988 (PDR OP 88-095, improper brazing material), and April 29, 1988 (PDR OP 88-094, improper brazing procedure), specifying corrective actions with completion verified as of May 27, 1988.

The corrective actions regarding the issuing of improper brazing material appeared to be satisfactory. However, the NRC inspectors noted that the weld data sheet (ADM08-300, Exhibit A) was initialed and dated by the weld engineer, without realizing that the incorrect brazing procedure was specified. The weld engineer routinely uses a desk-top procedure and checklist that are not part of the applicable administrative procedure. The NRC inspectors discussed the use of uncontrolled desk-top procedures and simple checklists with licensee personnel.

Additionally, the quality control review required to signify agreement with specified brazing requirements was initialed and dated on the weld data sheet without ensuring the correct brazing procedure was specified. After the inspectors discussed this matter with the licensee, an additional PDR was written on June 14, 1988 (PDR QC-88-011, QPS review of WR04105-87 failed to identify the wrong brazing procedure was to be used in the field). The PDR noted that a memorandum was written to all QC quality plant support personnel reminding them what type of review is required by Procedure QP 12.1, paragraph 7.1.2, and that QP 12.1 is to be revised to enhance review process to clearly state what documents are to be reviewed.

6.4.2 Results Summary

The NRC inspectors determined that the corrective actions taken by the licensee regarding the cooler repair appeared to be adequate. However, the NRC inspectors concluded that the final maintenance and QC review to determine technical adequacy and completeness of the package should have been completed before the shift supervisor restored the system to service.

7. EXIT INTERVIEW

The NRC inspectors held meetings with licensee supervisory and management personnel periodically during the course of the inspection to discuss the status of the inspection. The NRC inspectors met with the licensee's representatives (included in the list in Appendix A to this report) on June 17, 1988, to summarize the inspection scope and findings and the to discuss the observations and potential enforcement findings. Although proprietary material was reviewed during the inspection, no proprietary material is contained in this report.

APPENDIX A

PERSONS CONTACTED

Wolf Creek Nuclear Operating Corporation Personnel

*B. D. Withers, President
*F. T. Rhodes, Vice President Nuclear Operations
*R. M. Grant, Vice President Quality
*J. A. Bailey, Vice President Engineering and Technical Services
*G. D. Boyer, Plant Manager
*C. E. Parry, Manager-Quality Assurance
*G. W. Reeves, Manager-Quality Control
*W. M. Lindsay, Manager-Quality Evaluation
*R. H. Belote, Manager-Nuclear Safety Engineering
*J. M. Pippin, Manager-Nuclear Plant Engineering
*A. A. Freitag, Manager-Nuclear Plant Engineer-W.C.
*R. W. Holloway, Manager-Maintenance and Modification
*M. G. Williams, Manager-Plant Support
*O. L. Maynard, Manager-Licensing
*S. Wideman, Licensing
*K. Peterson, Supervisor-Licensing
*J. A. Zell, Manager-Training
*C. W. Fowler, Manager-I&C
*R. J. Potter, Manager-Material/Supplier Quality
*W. B. Wood, General Counsel
*J. L. Houghton, Supervisor Operations
*M. L. Johnson, Nuclear Coordinator-KG&E
*W. B. Norton, Supervisor Reactor Engineering
*L. Payne, Supervisor Quality Plant Support
*R. E. Gimple, Supervisor Materials Quality
*C. G. Patrick, Supervisor Quality Systems
*C. J. Hoch, Quality Assurance Technician
R. S. Benedict, Manager Plant Inspection
R. S. Robinson, Supervisor, I&C Maintenance
W. G. Eales, Jr., Manager Electrical Systems Engineering
N. Hoadley, Lead Engineer, Nuclear Plant Engineering
A. Clason, Manager Engineering Support
T. Deddens, Outage Manager
L. Stevens, Lead Engineer, Nuclear Plant Engineering

Other licensee employees contacted included operators, engineers, auditors, technicians, mechanics, and office personnel.

NRC Personnel

*F. C. Hawkins, Chief, Quality Operations Section, NRR
*P. W. O'Connor, Project Manager, NRR
*J. Jaudon, Deputy Director, Division of Reactor Safety, RIV
*B. Bartlett, Wolf Creek Senior Resident Inspector, RIV
M. E. Skow, Wolf Creek Resident Inspector, RIV
*B. Little, Callaway Senior Resident Inspector, RIII

*Denote those attending the exit meeting on June 17, 1988