UNITED STATES

October 8, 1999

J. H. Swailes, Vice President of
Nuclear Energy
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

SUBJECT: NRC INSPECTION REPORT NO. 50-298/99-03

Dear Mr. Swailes:

This refers to the inspection conducted on August 23 to 27 and September 7 to 10, 1999, at the Cooper Nuclear Station facility. The purpose of the inspection was to ensure that the licensee had effective programs for (1) identifying, evaluating, and correcting problems which could impact safe plant operations and (2) correcting and limiting non-compliance with NRC regulations. The enclosed report presents the results of this inspection.

On the basis of the sample reviewed, your corrective action program was generally implemented with an appropriate threshold for identifying and classifying adverse conditions. During the review of problem identification reports and other associated corrective action documents, the inspectors observed untimely long-term corrective actions associated with the operation of the residual heat removal system, which resulted in a operable but degraded condition for years. In addition, the logic testing associated with the residual heat removal system was an example of a failure to implement prompt and adequate corrective actions similar to issues identified in a previous noncited violation documented in NRC Inspection Report 50-298/99-04.

Based on the results of this inspection, the NRC has determined that two Severity Level IV violations of NRC requirements occurred. These violations are being treated as noncited violations, consistent with the Interim Enforcement Policy for pilot plants. These noncited violations are described in the subject inspection report. If you contest the violation or severity level of these noncited violations, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to the Regional Administrator, U.S. Nuclear Regulatory Commission, Region IV, 611 Ryan Plaza Drive, Suite 400, Arlington, Texas 76011, the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001; and the NRC Resident Inspector at the Cooper Nuclear Station facility.

In accordance with 10 CFR 2.790 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response, if requested, will be placed in the NRC Public Document Room.

Should you have any questions concerning this inspection, we will be pleased to discuss them with you.

Sincerely,

/s/

John L. Pellet, Chief
Operations Branch
Division of Reactor Safety

Docket No.: 50-298
License No.: DPR-46

Enclosures:
NRC Inspection Report No.
50-298/99-03

cc w/enclosures:
G. R. Horn, Senior Vice President
of Energy Supply
Nebraska Public Power District
1414 15th Street
Columbus, Nebraska 68601

John R. McPhail, General Counsel
Nebraska Public Power District
P.O. Box 499
Columbus, Nebraska 68602-0499

B. L. Houston, Nuclear Licensing
and Safety Manager
Nebraska Public Power District
P.O. Box 98
Brownville, Nebraska 68321

Dr. William D. Leech
Manager - Nuclear
MidAmerican Energy
907 Walnut Street
P.O. Box 657
Des Moines, Iowa 50303-0657

Ron Stoddard
Lincoln Electric System
1040 O Street
P.O. Box 80869
Lincoln, Nebraska 68501-0869

Michael J. Linder, Director
Nebraska Department of Environmental
Quality
P.O. Box 98922

Lincoln, Nebraska 68509-8922

Chairman
Nemaha County Board of Commissioners
Nemaha County Courthouse
1824 N Street
Auburn, Nebraska 68305

Cheryl K. Rogers, Program Manager
Nebraska Health and Human Services System
Division of Public Health Assurance
Consumer Services Section
301 Centennial Mall, South
P.O. Box 95007
Lincoln, Nebraska 68509-5007

Ronald A. Kucera, Director
of Intergovernmental Cooperation
Department of Natural Resources
P.O. Box 176
Jefferson City, Missouri 65102

Jerry Uhlmann, Director
State Emergency Management Agency
P.O. Box 116
Jefferson City, Missouri 65101

ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR 46
Report No.:	50-298/99-03
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	August 23 to 27, 1999 and September 7 to 10, 1999
Inspectors:	Paul C. Gage, Senior Reactor Engineer, Operations Branch Steve L. McCrory, Senior Reactor Engineer, Operations Branch Michael C. Hay, Resident Inspector, Project Branch C
Approved By:	John L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1: Supplemental Information

Attachment 2: Initial Material Requested

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report No. 50-298/99-03

This was an announced inspection to review the corrective action process, which included the methods used for identification, cause investigation, and correction of quality-related problems. The inspectors used recently written Inspection Procedure 71152, AIdentification and Resolution of Problems,@ to conduct the inspection.

Inspection findings were assessed according to potential risk significance and were assigned colors of GREEN, WHITE, YELLOW, or RED. GREEN findings are indicative of issues that, while not necessarily desirable, represent little risk to safety. WHITE findings would indicate issues with some increased risk to safety, which may require additional NRC inspections. YELLOW findings would be indicative of more serious issues with higher potential risk to safe performance and would require the NRC to take additional actions. RED findings represent an unacceptable loss of margin to safety and would result in the NRC taking significant actions that could include ordering the plant shut down. No individual finding by itself would be indicative of either acceptable or unacceptable performance. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

Mitigating Systems

$ Green: In using the cornerstone significance determination process, this issue was determined to have very low risk significance because the system remained operable, although degraded. The residual heat removal heat exchanger operator workaround conditions involving the operation of the heat exchanger outlet valves (including the service water side) had existed for several years. Long-term corrective actions to restore the system=s ability to maintain temperature control during shutdown cooling mode of operation, according to the system=s original design, had not been developed and implemented. Failure to establish prompt corrective actions for conditions adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation (50-298/9903-01), consistent with the Interim Enforcement Policy for pilot plants (Section 1RO7).

$ Green: In using the cornerstone significance determination process, this issue was determined to have very low risk significance because the system remained operable, although degraded. A narrowly focused approach in response to Generic Letter 96-01 involving surveillance issues associated with logic testing led to a recent noncited violation (50-298/9904-04) for inadequate corrective actions. Subsequent to the noncited violation, a condition, described in Licensee Event Report 99-005, addressed related circumstances associated with surveillance testing of the residual heat removal logic contacts. The corrective actions associated with the residual heat removal logic testing identified another example of the previously documented noncited violation (Section 1R22).

Occupational Radiation Safety

$ Green: In using the cornerstone significance determination process, this issue was determined to have very low risk significance because there was no unintended exposure or substantial potential for one and the ability to assess dose was not compromised. During withdrawal of the transverse incore probe from the reactor core, radiation levels exceeded the 5000 millirems per hour limit of the survey meter in use. As a result, on May 24, 1997, the extent of the radiation levels was unknown. The failure to perform an adequate radiological survey was a violation of 10 CFR Part 20, Section 1501. This violation is being treated as a noncited violation (50-298/9903-02), consistent the Interim Enforcement Policy for pilot plants (Section 2OS4).

Other

Problem Identification and Resolution Conclusion

The corrective action program was generally implemented adequately across all cornerstones, with very low risk significance examples of untimely corrective actions. The licensee=s self-assessments were appropriately focused on substantive performance improvement areas. Licensee management identified improving ownership, accountability, and support as a site-wide improvement area and was developing improvement plans at the end of the inspection.

Report Details

1. REACTOR SAFETY

1a. Initiating Events

1RO1 Adverse Weather

1. Inspection Scope

The inspectors reviewed the licensee=s disposition of industry operating experience contained in Information Notice 98-02, as part of the sample of responses to industry information. This was identified as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems.

2. Observations and Findings

Information Notice 98-02 addressed potential common-cause failure mechanisms of safety-related systems caused by extremely cold weather. The inspectors noted that the licensee appropriately screened the information for applicability to its facility. Applicable items were placed in the corrective action process for review to determine if the recommendations were implemented and concerns were addressed. The inspectors noted that a timely review was performed by the licensee, ensuring that its programs were adequate to address the cold weather concerns prior to the next cold weather period following the issuance of Information Notice 98-02.

1R12 Maintenance Rule Implementation

a. Inspection Scope

The inspectors reviewed the licensee=s maintenance rule implementation and maintenance preventable functional failure determinations associated with systems, structures, and components determined to be within the scope of the maintenance rule, as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems.

b Observations and Findings

Administrative Procedure 0.27, AMaintenance Rule Program,@ Revision 8, provided the guidance for implementation and compliance with the maintenance rule program at Cooper Nuclear Station. If any system, structure, or component performed such that unavailability was greater than 80 percent of the established unavailability criteria, or the number of identified failures were within one of exceeding the reliability criteria, then the affected function for the system, structure, or component was placed on the alert list. Functions placed on the alert list were part of the monthly maintenance rule report and were readily available to all levels of facility management.

Although system engineers were required to perform functional failure evaluations for conditions documented by a problem identification report, significant condition report, condition adverse to quality report, or resolve condition report, as applicable, there was no documented specification for when this evaluation was to be accomplished. Licensee representatives stated that management expected the functional failure evaluations to be performed within 30 days of the initial documentation, which identified the condition.

Significant Condition Report 98-0899 and Resolve Condition Report 99-0452 documented problems associated with prolonged delays in performing various aspects of evaluating documented problems for functional failures within the maintenance rule program. Significant Condition Report 98-0899, dated December 1, 1998, specified more than 3500 outstanding maintenance rule functional failure evaluations that had not been entered in the program database or completed, contrary to the management expectation above. All problem identification reports were sent from the corrective action group and the work control center to plant engineering.

Resolve Condition Report 99-0452, initiated on June 24, 1999, documented 469 problem identification reports that needed to be reviewed and dispositioned. Licensee personnel performed an electronic comparison between the nuclear action item tracking database and the maintenance rule database, which confirmed all problem identification reports were captured. Maintenance rule personnel reviewed the 469 problem identification reports for potential functional failures, and determined that approximately one-third (151) needed a functional failure evaluation performed by the responsible system engineers. The inspectors noted that none of the originally missed evaluations were classified as a significant condition report, which limited the significance of the problem. Corrective actions, as documented in the respective condition reports, included: (1) maintenance rule training for system engineers, (2) plant engineering supervisors- re-emphasis of management for timeliness expectations, and (3) clarification or expectations and responsibilities within applicable maintenance rule program procedures. The inspectors observed that the proposed procedure changes were still in the implementation process awaiting review and approval and, therefore, could not be assessed during the inspection.

During the onsite inspection effort, information contained within the maintenance database indicated 153 problem identification reports awaited the performance of a functional failure determination. The inspectors noted that approximately 30 percent (46 of 153) of the problem identification reports exceeded the stated 30-day expectation for completing the functional failure evaluation. The inspectors observed that three systems (diesel generator, neutron monitoring instrumentation, and reactor equipment cooling) each had problem identification reports that exceeded 300 days without completing the functional failure evaluation.

An extended delay evaluating potential functional failures delayed the corresponding of determination cause and corrective actions. For multiple failures, this could delay placing the affected system, structure, or component on the alert list, or evaluate for Category (a)(1) status in accordance with the maintenance rule. The inspectors reviewed the past 18-months= data of the three systems above. The established performance criteria would not have been exceeded had all of the incomplete evaluations been functional failures.

During the review of plant events within the past 12 months, the inspectors evaluated the maintenance rule application regarding the failure of a reactor core isolation cooling (RCIC) system motor-operated valve on April 14, 1999, when the licensee declared the RCIC system inoperable due to a failure of RCIC Cooling Valve RCIC-MO-132. As the operator placed the valve control switch to the closed position in order to restore the normal standby configuration, Valve RCIC-MO-132 failed to reposition. Failure of the valve to reposition was documented in Resolve Condition Report 99-0311. Licensee personnel attributed the cause of valve=s failure to reposition to dirt on the contacts of the close torque switch. Licensee personnel concluded that since the close torque switch was not used when opening the valve, the valve was capable of performing its intended safety function to open, and that the documented condition was not a functional failure. The inspectors noted that no previously identified similar problems had occurred. The inspectors concluded that the functional failure determination was appropriate for the failed RCIC motor-operated valve.

1R14 Nonroutine Plant Evolutions

a. Inspection Scope

As part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems, the inspectors reviewed the 250 and 125 Vdc Class 1E distribution systems. This was part of the selected sample from risk significant systems determined from the plant=s probabilistic safety analysis and included recent Class 1E batteries= multiple cell failures.

b Observations and Findings

The inspectors noted that the Class 1E dc distribution systems were divided into seven individual functions, which included three battery charger functions, two battery functions, and two distribution functions, as documented by the expert panel meeting minutes of March 30, 1999. The probabilistic safety analysis supported the classification of the battery chargers as non-risk significant systems with no criterion needed for unavailability, and a performance criterion of two functional failures over a 18-month moving window for monitoring system reliability. The batteries and distribution systems were classified as risk significant systems with established performance criteria of no functional failures and 0.5 percent unavailability.

Regulatory Guide 1.160, AMonitoring the Effectiveness of Maintenance at Nuclear Power Plants,@ Revision 2, documented, in part, that for some systems, structures, and components, the maintenance preventable functional failure performance criterion may

be too small to be effectively monitored and trended, as required by the rule. In these cases, the licensee should establish performance or condition monitoring criteria that can be monitored and trended so that the licensee can demonstrate that maintenance is effective. The inspectors determined that the batteries and distribution systems were appropriately monitored for performance through the weekly and quarterly surveillances required by the plant=s technical specifications.

On July 7, 1999, during the quarterly surveillance of the 125 volt AA@ battery, a single cell (No. 7) failed the individual cell voltage Technical Specification 3.8.6 limit of greater than or equal to 2.13 Vdc (Problem Identification Report 4-02942). The licensee staff placed the cell on an individual equalizing charge and appropriately increased the surveillance frequency in accordance with Technical Specification 3.8.6. Seven days later (July 14, 1999), the affected cell=s individual cell voltage was rechecked and found to be 2.08, which was less than the Category C limit of 2.10 identified in technical specifications. As a result, the 125 volt AA@ battery was declared inoperable and Cell 7 was replaced. Immediate visual inspections identified 21 additional battery cells distributed throughout all four Class 1E batteries, as showing various degrees of positive plate growth or large flake formations. The inspectors noted that all of these 21 additional battery cells passed their quarterly surveillance tests. The inspectors noted that all of the cells, which had experienced either the abnormal growth or large flake formations on the positive plates had been replaced, and that visual checks of all cells had been included as part of the weekly surveillance activity. Subsequent analysis of the positive plates indicated an excessive amount of calcium existed compared to the manufacturer=s design specification. The inspectors noted that long-term corrective actions were being developed and awaiting final approval from the corrective action review board.

Initiating Events Conclusion

No findings. However, the inspectors found that several hundred problem identification reports were not being evaluated for functional failures within the licensee management=s expectation. Three systems (diesel generator, neutron monitoring instrumentation, and reactor equipment cooling) had examples of problem identification reports that exceeded 300 days without completing a functional failure determination for the associated condition. The inspectors determined that the inclusion of the potential failures would not have exceeded the corresponding performance criteria, and that no additional regulatory actions required by the maintenance rule were necessary.

1b. Mitigating Systems

1RO7 Heat Sink Performance

a. Inspection Scope

As part of the programmatic evaluation of the effectiveness of the licensees identification and resolution of problems, the inspectors reviewed the residual heat removal (RHR) system. The RHR system was included as part of the sample of systems designated in the plant=s probabilistic safety analysis as risk significant. Additionally, the RHR system had been the subject of several inspection findings during the previous year (NRC

Inspection Reports 50-298/98-02, 98-03, 98-04, 98-05, 98-07, 98-08, 98-09, 98-15, 98-22, and 99-05).

b Observations and Findings

The RHR system inspection focused on two long-standing issues:

$ Design features leading to excessive cooldown while operating in the shutdown cooling mode with low river water temperatures and

$ Leakage through the RHR heat exchanger service water outlet Valves SW-MO-89 A(B).

In 1986, operations personnel initiated a request to revise operating procedures to allow throttling of Valves SW-MO-89 A(B) to control cooldown rate when operating in the shutdown cooling mode of RHR. In 1988, the licensee performed a 10 CFR 50.59 review to assess the use of Valves RHR-MO-12 A(B), the RHR Heat Exchanger A(B) outlet valve, as a throttle valve to aid in controlling reactor cooldown in the shutdown cooling mode of RHR instead of Valves SW-MO-89 A(B). The review concluded that using Valves RHR-MO-12 A(B) did not affect the emergency core cooling function nor engineered safety features of the RHR system. The review also concluded that Valves RHR-MO-12 A(B), a 16-inch, 300 lb motor-operated gate valve manufactured by Anchor/Darling, would not be adversely affected by being operated in a throttled position during the shutdown cooling mode of RHR. The inspectors reviewed the 10 CFR 50.59 evaluation and concurred with the licensee=s conclusions.

In early 1998, after 10 years in use, the licensee concluded that the operation of Valves RHR-MO-12 A(B) as described above constituted an operator workaround and initiated Problem Identification Report 2-27259, on March 26, 1998. This requested that a design modification be developed that would allow Valves RHR-MO-12 A(B) to be remotely throttled from the control room. The licensee initiated Change Evaluation Document 1998-0268, to develop the design modification requested by operations. At the time of the inspection, the licensee had not approved Change Evaluation Document 1998-0268. However, the licensee had established a due date of December 1, 1999, for completion of the design package, with installation scheduled before the start of the next outage in 2000.

The modification proposed was to replace the two-position switches on the control room and auxiliary shutdown panels with three-position switches that included a Astop@ position. The switches were the same with regard to design and performance specifications with the exception of the Astop@ switch position. The installation basically involved only replacement of the switch assemblies.

During the course of the inspection, the licensee was unable to identify any documentation that specifically described the problem regarding RHR Heat Exchanger A(B) or any engineering evaluations that assessed the problem and possible resolutions. During discussions with the inspectors, the system engineers for the RHR

system stated that the RHR system had been inadequately designed to account for low river water temperature while operating in the shutdown cooling mode. Therefore, the licensee relied on this operator workaround for this condition. As a result of the inspection, the licensee initiated Change Evaluation Document 4-04053 to investigate and document the RHR design inadequacy.

Between 1986 and 1993, the licensee revised Procedure SOP 2.2.70, ARHR Service Water Booster Pump System,@ to permit windmilling (flow through an idle pump allowing the impeller to freely rotate) the booster pumps while operating in the shutdown cooling mode of RHR. Since the booster pumps were originally intended to maintain pressure on the service water side of the RHR heat exchanger greater than the RHR side, the licensee included in the revised procedure the requirement that the reactor be in Mode 4 or 5 before windmilling the booster pumps.

The licensee initiated the change primarily to reduce erosion of the RHR heat exchanger Service Water Outlet Valves SW-MO-89 A(B). The large amount of sand in the river water and high velocities across the valve seating and sealing surfaces when the booster pumps were running with the valves in a throttled position produced the erosion. That led to leakage through the valves when the system was shutdown during normal power operations. Windmilling the pumps greatly reduced the erosive effects.

A consequence of the service water flow through the RHR Heat Exchangers A(B) due to valve leakage was increased biological activity that resulted in tube fouling on the service water side of RHR Heat Exchangers A(B). The process occurred at the bacterial level and resulted in the deposit of a tenacious layer of silt and bacterial waste that eventually plugged the tubes. The extent to which the fouling could progress was observed in RHR Heat Exchanger B when it was opened for cleaning and inspection during the 1997 refueling outage. When the impact of the fouling was analyzed, RHR Heat Exchanger B retained the required heat removal capability for worst-case accident conditions, but was significantly degraded.

While investigating the RHR Heat Exchanger B condition observed in 1997, the licensee determined that the heat exchanger had been used for an extended period of time during an unscheduled outage (approximately 9 months) in 1994. During that outage the operators operated the service water side with the booster pumps windmilling.

According to comments made by a member of the licensee=s staff during a meeting with inspectors on August 26, 1999, in 1994, the operators used Valves SW-MO-89A(B) to control reactor temperature from the control panel in lieu of using Valves RHR-MO-12 A(B), which required coordination with an operator stationed at the valve-motor breaker. The operators progressively reduced the service water flow rate to approximately 800 gpm. Since the operators had not been permitted to windmill the RHR service water booster pumps since 1995, there had been no recurrence of throttling service water flow to control reactor temperature while operating RHR in the shutdown cooling mode.

As previously noted, the licensee had experienced leakage through Valves SW-MO-89 A(B) since some time prior to 1993. During those years, the

licensee made modifications to the valve internals, including the valve trim mechanism and coating materials. These efforts only extended the operational interval before the onset of leakage after repair from about 20 to approximately 150 hours. The licensee responded to the condition with monitoring of the leak rate through the valves and regular repair of the valves when the leakage became excessive. During much of that time, the license did not identify the link between the leakage of Valves SW-MO-89 A(B) and the fouling mechanism occurring in the RHR heat exchangers. Therefore, when the licensee observed significant fouling in RHR Heat Exchanger B in 1997, the principal corrective action implemented at that time was to ensure that both heat exchangers were cleaned during each refueling outage. This was because the licensee initially concluded that fouling buildup was an expected consequence of normal system use.

In late 1998, the licensee linked the RHR Heat Exchangers A(B) fouling mechanism (previously described) to low service water flow through the heat exchangers due to leakage through Valves SW-MO-89 A(B). The licensee reopened Significant Condition Report 98-0742 on February 11, 1999, and implemented an interim corrective action as follows to eliminate the fouling buildup in the RHR heat exchangers. The licensee performed a weekly preventative maintenance to operate the RHR service water booster pumps for short durations at a flow rate of approximately 4000 gpm. Analysis demonstrated that virtually all fouling buildup could be eliminated by the frequent operation of the system at normal operating flow rates (3500-4000 gpm).

The licensee contracted for additional engineering consulting services to provide recommendations for a permanent resolution of Valves SW-MO-89 A(B) leakage problem. Proto-Power Corporation provided the licensee with Conceptual Design Document, AEvaluation of Service Water Flow Through Valves SW-MO-89A/B,@ Revision 0, on July 23, 1999. The evaluation provided several recommendations, which included the installation of a DRAG7 type globe valve manufactured by Control Components, Inc., The evaluation also noted that these types of valves had been used at another facility as RHR service water isolation valves for the past 5 years with no problems. Subsequently, the licensee issued Contract Number 99-32, AService Water Booster System Valves,@ which solicited sealed bids by September 1, 1999, for a Control Components, Inc., DRAG7 type globe valve or equivalent to be installed during the refueling outage in 2000.

The inspectors concluded that the licensee began a process to modify the design and operation of the RHR system and its associated RHR service water booster system without fully understanding the design deficiencies and operational impacts. The licensee did not recognize a cause-and-effect relationship between RHR excessive heat removal, in the shutdown cooling mode throttling service water flow through the heat exchanger and heat exchanger fouling rate. Once the excessive fouling of RHR Heat Exchanger B was observed, the licensee took nearly 2 years to understand the fouling mechanisms at work in the heat exchangers. The licensee applied various interim or stop-gap measures during the last 10 years regarding the RHR design deficiencies that led to excessive cooldown under certain conditions and the Valves SW-MO-89 A(B) leakage that exacerbated fouling mechanisms in the RHR heat exchangers. Since this

condition resulted in serious and ongoing system degradation, but did result in inoperable systems, the risk significance and safety impact was very low. However, failure to establish timely corrective actions for conditions adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. Licensee personnel initiated Resolve Condition Report 99-0644 to enter and track this issue within the corrective action program. This Severity Level IV violation is being treated as a noncited violation, consistent with the Interim Enforcement Policy for pilot plants (50-298/9903-01).

1R22 Surveillance Testing

a. Inspection Scope

As part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems, this inspection included a review of licensee identified problems involving missed technical specification surveillances. This encompassed a review of condition reports, licensee event reports, and noncited violations to determine the licensee=s effectiveness to identify, evaluate, correct, and report conditions in non-compliance with NRC regulations. This review included Significant Condition Reports 98-0268, 98-0285, and 99-0399.

b. Observations and Findings

Missed Technical Specification Surveillances Associated with the Testing of Logic Circuits

Significant Condition Report 98-0268, described that on April 5, 1998, the licensee identified the average power range monitor system high flux (fixed) reactor protection system trip setpoint was not being functionally tested as required by Technical Specification 4.1.A. The licensee determined that the root cause of the missed functional testing was due to a lack of understanding of the average power range monitoring circuitry, which resulted in a deficient surveillance procedure. This event was appropriately reported to the NRC by Licensee Event Report (LER) 1998-005. Licensee Event Reports 1998-005 was closed in NRC Inspection Report 50-298/98-008 as a violation of Technical Specification 4.1.A.
Significant Condition Report 98-0285, addressed that on April 6, 1999, the licensee identified that four of eight reactor protection scram relays were not tested as required by Technical Specification 3.0.3. The licensee determined that the root cause of the failure to test the relays was the result of inadequate translation of testing requirements into surveillance procedures prior to initial plant operation. This event was appropriately reported to the NRC by LER 1999-002. Licensee Event Report 1999-002 was closed in NRC Inspection Report 50-298/99-04 as a violation of 10 CFR Part 50, Appendix B, Criterion XVI, based on the licensee having failed to implement appropriate corrective actions to ensure that surveillance testing of logical circuits met technical specification requirements.

Significant Condition Report 99-0399, described that on June 1, 1999, the licensee identified that two residual heat removal system logic relays were not adequately tested

to satisfy logic system functional test requirements for the reactor vessel shroud level function as required by Technical Specification Surveillance Requirement 3.3.5.1.5, Function 2.e. The reactor vessel shroud level function served as both a permissive for, and an inhibit to, operator actions which divert residual heat removal system low pressure coolant injection flow from the reactor vessel to the containment for drywell spray, torus spray, or torus cooling water. On June 3, 1999, during further review, the licensee identified that contacts for override keylock switches were not verified open to ensure that test results were of the relay contacts and not the key-locked switches due to the switches being in parallel with the relay contacts. This event was appropriately reported to the NRC by LER 1999-005.

The inspectors determined that immediate corrective actions for these significant condition reports were appropriate and prevented a recurrence of the specific problem. However, the inspectors noted that the licensee had prior opportunities to identify and correct these inadequate surveillance procedures. As previously discussed in NRC Inspection Report 50-298/99-04, the licensee had the following prior opportunities to identify and correct the surveillance procedures. In May 1994, inspectors identified failures to properly test relay contacts on essential electrical distribution load shed circuitry. This resulted in a plant shutdown required by technical specifications. Subsequent licensee responses to Confirmatory Action Letters NLS 940001, "Response to Confirmatory Action Letter," dated July 28, 1994, and NLS 9400026, "Response to Request for Additional Information," dated August 8, 1994, response to Escalated Enforcement Actions 50-298/94016-01 and -02, and NPPD Letter N65950028, "Reply to a Notice of Violation and Proposed Imposition of Civil Penalties," dated January 18, 1995, stated that the licensee would review all surveillance testing requirements and conduct functional testing to ensure surveillance testing met technical specification requirements. Testing of logic systems was also addressed in Generic Letter 96-01. The licensee responded to the letter in April 1996, indicating that a complete review of logic tests had been performed prior to the letter and no further review was required.

The failure to identify and correct inadequate relay testing during the 1994 and 1995 surveillance test validation program and other opportunities was another example of a violation identified in NRC Inspection Report 50-298/9904-04. The licensee=s corrective actions for the previously documented noncited violation identified this additional example. The safety significance of this failure to perform surveillance tests of the two residual heat removal system logic relays was very low, since the contacts in question passed their subsequent functional test and the failure of these contacts would not have caused a loss of the emergency core cooling systems= ability to provide sufficient flow for removal of decay heat.

The inspectors reviewed the licensee=s long-term corrective actions for the three significant condition reports and noted that the long-term corrective actions for Significant Condition Report 98-0268 and 98-0285 appeared reasonable; however, they were narrowly focused. The inspectors were informed that one reason the corrective actions were narrowly focused was that the same individual had performed the surveillance validation in 1994 for both issues resulting in Significant Condition Reports 98-0268 and 98-0285. For this reason, the licensee concentrated its corrective actions on those

systems the individual validated. The inspectors noted that Significant Condition Report 99-0399 contained long-term corrective actions that included a more generic review of the surveillance procedures to confirm that they acceptably demonstrated logic system functional testing surveillance requirements.

Overall, the licensee was effectively identifying problems associated with surveillance procedures that did not effectively test logic systems to satisfy technical specification surveillance requirements. The inspectors noted that the licensee=s past effort to correct these types of problems did not prevent recurrence resulting in a second example of a noncited violation previously identified in NRC Inspection Report 50-298/99-04.

Missed Technical Specification Surveillances Associated with Changing Operational Conditions

The inspectors reviewed two noncited violations from previous NRC inspections, which were not previously reviewed by the NRC, to determine if the violations were entered into the corrective action program and if they were resolved or being resolved in a timely manner commensurate with their safety significance.

Noncited Violations 50-298/98004-04 (addressed by Significant Condition Report 98-0214) and 50-298/98002-04 (addressed also by Significant Condition Report 98-0214) were found to be entered into the corrective action program. The inspectors noted that both violations were a result of the licensee failing to perform technical specification surveillances prior to operational mode changes. The inspectors noted that Significant Condition Report 98-0214 identified that corrective actions associated with Condition Report 97-1075 were not effective to prevent a recurrence.

Condition Report 97-1075 contained a corrective action to completely review the technical specifications to identify all operability verifications required prior to mode change to startup and hot standby or run. These corrective actions were completed in 1997, but were not sufficient to identify the two examples resulting in Noncited Violations 50-298/98004-04 and 50-298/98002-04. Therefore, corrective actions associated with Condition Report 97-1075 were ineffective. However, the two noncited violations occurred in December 1995 and May 1997 and the corrective actions for Condition Report 97-1075 were not completed until August 1997. The inspectors determined that even though the corrective actions were not sufficient to prevent repetition, since they were not complete at the time of these two violations, they did not result in a repetitive significant condition adverse to quality, and no violation of 10 CFR Part 50, Appendix B, Criterion XVI, occurred.

Mitigating Systems Conclusion

Residual heat removal heat exchanger operator workaround conditions involving the operation of the heat exchanger outlet valves (including the service water side) had existed for several years. Long-term corrective actions to restore the systems ability to maintain temperature control during shutdown cooling mode of operation according to the system=s original design, had not been developed and implemented. Failure to

establish timely corrective actions for conditions adverse to quality was a violation of 10 CFR Part 50, Appendix B, Criterion XVI. This violation is being treated as a noncited violation (50-298/9903-01), consistent with the Interim Enforcement Policy for pilot plants.

A narrowly focused approach in response to Generic Letter 96-01 involving surveillance issues associated with logic testing led to a recent noncited violation (50-298/9904-04) for inadequate corrective actions. Subsequent to the noncited violation, a condition, described in Licensee Event Report 99-005, addressed related circumstances associated with surveillance testing of the residual heat removal logic contacts. The corrective actions associated with the residual heat removal logic testing identified another example of the previously documented noncited violation.

1c. Barrier Integrity

1RO7 Heat Sink Performance

a. Inspection Scope

As part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems, the inspectors reviewed the safety relief valve and automatic depressurization systems as part of the sample of systems designated by the probabilistic safety analysis as risk significant. Additionally, the automatic depressurization/safety relief valve systems provided information to evaluate the barrier integrity cornerstone.

b. Observations and Findings

The inspectors interviewed the system engineers for automatic depressurization/safety relief valve systems and reviewed Significant Condition Report 99-0346, AFailure of Safety Relief Valve Serial Number 387, to Meet TS ± 3% Auto Actuation Tolerance.@

The automatic depressurization/safety relief valve systems had few problems over the past year. The most significant problem was that one of the safety relief valves removed for testing following the October 1998 refueling outage lifted above the setpoint tolerance band. This was noteworthy because the licensee had modified the pilot discs in the Target Rock safety relief valves. The modification consisted of replacing the discs made of Stellite J 6 material with ones made of Stellite J 21. During a mid-cycle outage in March 1998, all eight safety relief valves were replaced with ones containing the Stellite J 21 material. Target Rock safety relief valves had exhibited corrosion bonding problems that led to high setpoint drift industry wide. Although there was no industry consensus, Cooper Nuclear Station elected to replace the existing pilot discs with ones made of Stellite J 21 based on the success experienced at another similar designed nuclear power plant facility. At about the same time, the licensee converted to the improved technical specifications, which relaxed the tolerances on safety relief valve setpoints from ± 1 percent to ± 3 percent. In a subsequent retest of the valves, all but one (safety relief valve Serial Number 387) were within the relaxed setpoint tolerances. The licensee determined that the failure of safety relief valve Serial No. 387 was

anomalous rather than a repeat failure. This was because the setpoint drift error was greater than two standard deviations from the mean setpoint drift for valves containing Stellite J 21. The inspectors agreed with the licensee=s assessment and concluded that the licensee=s corrective actions for the safety relief valve were appropriate.

1R22 Surveillance Testing

a. Inspection Scope

The inspectors reviewed selected licensee identified conditions adverse to quality associated with primary and secondary containment isolation valves and reactor building to suppression chamber vacuum breakers. These conditions were reviewed to determine the licensee=s effectiveness in implementing the corrective action process associated with the barrier integrity cornerstone.

b Observations and Findings

The inspectors reviewed Significant Condition Report 98-1235 pertaining to a secondary containment isolation valve that was not verified closed as required by Improved Technical Specification Surveillance Requirement 3.6.4.2.1. This condition was appropriately reported to the NRC by LER 1998-013. Licensee Event Report 1998-013 was identified in NRC Inspection Report 50-298/99-01 as a violation of Technical Specification Surveillance Requirement 3.6.4.2.1. The licensee determined that inadequate review of drawing changes resulted in Valve RW-V-1258 not being identified and placed into the surveillance procedure for implementing the improved technical specifications. The inspectors reviewed the licensee=s corrective actions and found them acceptable. Corrective actions associated with the event led to the licensee identifying an additional potential component omission. However, the root cause was determined to be unrelated to the inadequate drawing change review. The licensee appropriately initiated Problem Identification Report 4-00619 to assess the finding. Subsequently, the potential component was determined to not be a secondary containment isolation valve. During the review of the associated problem identification report, the inspectors found the licensee=s evaluation to be appropriate.

Significant Condition Adverse to Quality Report 98-0460 pertained to the failure to test reactor building suppression chamber vacuum breaker assemblies in accordance with Technical Specification 4.7.A.3.a was reviewed. This condition was appropriately reported to the NRC in LER 1998-008 and was identified in NRC Inspection Report 50-298/98-08 as a violation of Technical Specification 4.7.A.3.a. The licensee took appropriate immediate corrective actions and tested the associated vacuum breaker assemblies. Long-term corrective actions to prevent recurrence were reviewed and determined to be appropriate.

Barrier Integrity Conclusion

No findings.

1d. Emergency Preparedness

1EP1 Drill, Exercise, and Actual Events

3. Inspection Scope

As part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems, the inspectors reviewed the licensee=s response to Emergency Preparedness Exercise Weakness 9816-01. The weakness identified a failure of personnel entering the technical support and operations support centers to consistently perform contamination monitoring at the contamination control point.

4. Observations and Findings

The licensee, in Condition Adverse to Quality 98-0401, determined that the root cause was a failure to adequately communicate the requirement to perform contamination monitoring at the entrance to the technical support center and operations support center, once the control point was set up, regardless of whether response personnel had been monitored during exiting of the radiologically controlled area. The inspectors noted that no exercise had been conducted pursuant the documentation of the contamination monitoring issue. The licensee=s corrective actions included:

- An electronic mail to emergency response organization personnel discussing the weakness and expected performance,
- A revision to the chemistry/radiation protection coordinator position instruction manual to emphasize the requirement,
- A read-and-sign memorandum regarding monitoring requirements and expectations, and
- A revision to the lesson material for emergency response organization training emphasizing the requirements.

The inspectors reviewed the signature sheets that confirmed that the read-and-sign material had been seen by the emergency response organization members, and that the entry to the radiologically controlled area had a posted sign. Additionally, the inspectors confirmed the revisions to the position instruction manual and emergency response organization training material.

1EP3 Emergency Response Organization Augmentation

a. Inspection Scope

The inspectors reviewed the licensee=s disposition of industry operating experience contained in Information Notice 98-20, AProblems with Emergency Preparedness Respiratory Protection Programs,@ as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems.

b Observations and Findings

Information Notice 98-20 addressed problems associated with the emergency preparedness respiratory protection programs. The inspectors noted that the licensee appropriately screened the information for applicability to their facility. Applicable items were placed in the corrective action process for review to determine if the recommendations were implemented and concerns were addressed. The inspectors noted that the licensee effectively utilized Information Notice 98-20 and identified problems associated with emergency response organization personnel who were respiratory protection qualified and not keeping a clean shaven face required for respirator use. To address this problem, a site-wide Aclean shaven@ policy was implemented for those respiratory protection qualified personnel who were members of the emergency response organization.

Emergency Preparedness Conclusion

No findings.

2. RADIATION SAFETY

2a. Public Radiation Safety

2PS1 Gaseous and Liquid Effluent

5. Inspection Scope

The inspectors reviewed Condition Adverse to Quality Report 98-0431, as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems.

6. Observations and Findings

Condition Adverse to Quality Report 98-0431 was initiated based on an NRC inspection identifying that the licensee failed to include an explanation in the annual radioactive materials release report as to why the service water effluent radiation monitor was not repaired in a timely manner as required by technical specifications. The licensee determined the root cause to be personnel error since the person responsible for the limiting condition for operation tracking system failed to utilize the system properly. Corrective actions consisted of discussion with the individual responsible for the error and enhancements to the operating instructions associated with the limiting condition for operation index system. The inspectors found the corrective actions acceptable and identified no similar occurrences.

Public Radiation Safety Conclusion

No findings.

2b. Occupational Radiation Safety

2OS1 Access Control

7. Inspection Scope

The inspectors reviewed the licensee=s disposition of industry operating experience contained in Information Notice 99-04 pertaining to unplanned radiation exposures to radiographers, as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems.

8. Observations and Findings

Information Notice 99-04 addressed unplanned radiation exposures to radiographers. The inspectors noted that the licensee appropriately screened the information for applicability to its facility. Applicable items were placed in the corrective action process for review to determine if the recommendations were implemented and concerns were addressed. Although no examples of unplanned exposures were identified during the licensee=s historical review, the items addressed by Information Notice 99-04 were applied within the facility=s training program.

2OS4 Radiation Worker Performance

a. Inspection Scope

The inspectors reviewed selected licensee identified conditions adverse to quality related to the occupational exposure cornerstone.

b Observations and Findings

Significant Condition Report 98-0611 pertaining to an unplanned airborne radioactivity area being created while removing drywall head bolts was reviewed. The cause of the unplanned radioactive airborne activity was determined by the licensee to entail a lack of supervisory oversight and radiological department interface to ensure that proper radiological controls were implemented. NRC Inspection Report 50-298/98007-12 reviewed this event and determined that a violation of NRC requirements occurred involving the failure to follow radiation work permit requirements. This violation was treated as a noncited violation consistent with Section VII.B.1 of the NRC Enforcement Policy. The inspectors reviewed the licensee=s immediate and long-term corrective actions and determined they were technically adequate. The inspectors noted no similar occurrences had been identified by the licensee or NRC.

The inspectors reviewed Condition Adverse to Quality Report 98-0092. This report was initiated based on a review of past problem identification reports and radiation protection self assessments indicating that a generic trend existed related to documentation errors. The inspectors noted that the licensee performed a detailed review of all 1997 radiation protection job packages. The inspectors noted that the licensee effectively audited 189 radiation protection job packages and identified 305 discrepancies. The inspectors noted that corrective actions associated with documentation errors were effectively implemented. The licensee performed a review of all radiation protection job packages performed in the first quarter of 1998 and identified no discrepancies.

Condition Adverse to Quality Report 98-308 was reviewed. This report, initiated on May 5, 1998, documented multiple errors during the replacement of a transverse incore probe nearly a year earlier (May 24, 1997). These errors were found during an historical documentation review performed by licensee personnel. The identified errors included the failure to use a job package, an unplanned radioactive airborne condition occurred without evaluating the use respiratory protection or engineering controls to minimize personnel exposure, and use of a portable radiation survey meter that was not capable of evaluating the radiation exposure encountered. The inspectors noted that although multiple problems were encountered during the transverse incore probe replacement, no actions were taken to enter these deficiencies into the corrective action process following the job.
The inspectors were informed that the problems encountered during the job were not placed into the corrective action process due to none of the workers being aware that any abnormal conditions existed. The licensee addressed these deficiencies by including this event in lessons learned training that was received by radiation protection personnel. The inspectors identified no similar occurrences of radiation protection personnel not effectively utilizing the corrective action process.

During review of Condition Adverse to Quality Report 98-0308, the inspectors noted that one of the errors identified by the licensee was that the radiation protection technician providing job coverage for workers used a survey meter that had a maximum scale of 5000 millirems per hour. During the replacement of the transverse incore probe, the radiation levels in the work area exceeded 5000 millirems per hour. As a result, the radiation protection technician was unable to determine the magnitude and extent of radiation levels.

The inspectors interviewed the radiation protection technician providing job coverage for the transverse incore probe replacement. The radiation protection technician stated that during the transverse incore probe withdrawal from the reactor core, the portable radiation survey meter that was being used to access the radiological conditions was placed next to the shielded container into which the probe was being reeled. Subsequently, when the probe was withdrawn out of the reactor core the technician=s portable survey meter was off scale at 5000 millirems per hour for approximately 10 seconds until the probe was placed in the shielded container. The technician stated that the workers performing the activity were 6 to 10 feet away from the radiation source during the withdrawal of the probe and that no electronic dosimeters alarmed during the activity. The inspectors reviewed the electronic dosimetry records and verified that personnel exposures were monitored and normal for this type of activity. Therefore, the error did not lead to unintended exposure or, given the duration, substantial potential for one.

10 CFR 20.1501 requires that each licensee make or cause to be made surveys that may be necessary for the licensee to comply with the regulations in Part 20 and that are reasonable under the circumstances to evaluate the extent of radiation levels, concentrations or quantities of radioactive materials, and the potential radiological hazards that could be present.

Pursuant to 10 CFR 20.1003, *survey* means an evaluation of the radiological conditions and potential hazards incident to the production, use, transfer, release, disposal, or presence of radioactive material or other sources of radiation.

10 CFR 20.120 (a)(1)(i) states, in part, AThe licensee shall control the occupational dose to individual adults to the annual total effective dose equivalent limit of 5 rems.

The failure to make surveys to assure compliance with 10 CFR 20.1201(a)(1)(i) on May 24, 1997, during replacement of the transverse incore probe was identified as a violation of 10 CFR 20.1501. This Severity Level IV violation is being treated as a noncited violation, consistent with the Interim Enforcement Policy for pilot plants. This violation is in the licensee=s corrective action program as Condition Adverse to Quality Report 98-0308 (50-298/9903-02).

Occupational Radiation Safety Conclusion

During withdrawal of the transverse incore probe from the reactor core, radiation levels exceeded the 5000 millirems per hour limit of the survey meter in use. As a result, on May 24, 1997, the extent of the radiation levels was unknown. The failure to perform an adequate radiological survey was a violation of 10 CFR Part 20, Section 1501. This violation is being treated as a noncited violation (50-298/9903-02), consistent with the Interim Enforcement Policy for pilot plants.

3. PHYSICAL PROTECTION

3PP3 Response to Contingency Events

a. Inspection Scope

The inspectors reviewed the licensee=s disposition of industry operating experience contained in Information Notice 98-35, as part of the programmatic evaluation of the effectiveness of the licensee=s identification and resolution of problems..

b Observations and Findings

Information Notice 98-35 addressed threat assessments and consideration of heightened physical protection measures. The inspectors noted that the licensee appropriately screened the information for applicability to their facility. Applicable items were placed in the corrective action process for review to determine if the recommendations were implemented and concerns were addressed. The inspectors noted that the licensee made enhancements to their security procedures in response to the recommendations contained in Information Notice 98-35.

Physical Protection Conclusion

No findings.

4. OTHER ACTIVITIES

4OA1 Identification and Resolution of Problems

1. Inspection Scope

The inspectors reviewed 10 quality assurance audits and licensee self-assessments regarding the corrective action program separately from the cornerstone corrective action issues previously discussed. Additionally, the inspectors reviewed 10 condition reports, listed in Attachment 1, that each aggregated a number of related issues.

2. Observations and Findings

Quality Assurance Audits and Self Assessments

The licensee=s quality assurance audits and self-assessments candidly and critically reported problems with the corrective action program and specific root-cause investigations and corrective actions. The audits and assessments identified weak or inadequate root-cause determination, weak, ineffective, or inappropriate corrective actions, and weak use or non-use of industry experience, especially NRC information. The audits cited that the corrective actions often did not address the root cause and that there were significant instances of repetitive problems that were not recognized or not effectively evaluated. For example, the June 1999 Quality Assurance Audit contained the following findings with respect to the licensee=s program:

$ Strong problem identification was observed;

$ Adequate procedures and electronic process existed;

$ There was a need to raise priority for evaluation and resolution for non-safety related, risk-significant non-conformances;

$ The corrective action program implementation needed improvement overall with emphasis on: (1) development of problem statements, (2) use of (diverse) root-cause techniques, (3) assignment of interim corrective actions (w/governing process), and (4) develop specific corrective actions (flow directly from root cause);

$ Generally effective evaluation of industry events was observed, but evaluations routinely omitted NRC experience;

$ Failure to explain why previous corrective actions were ineffective was observed (procedure non-compliance); and

$ Adequate corrective actions were taken with regard to Audit 98-17 and Surveillance S408-9902.

These audits led to the ACooper Nuclear Station Business Plan Initiative - Improve The Corrective Action Program,@ initiated on July 21, 1999. This replaced the former corrective action program performance improvement plan that had progressed through Revision G, February 5, 1999. At the time of the inspection, the licensee not yet issued

many of the new or revised implementing procedures; although, several were in final draft form.

The inspectors noted improvement in the corrective action process exemplified by recent aggregation of related problems suggesting generic issues and the development of the monthly corrective action trend report.

The inspectors reviewed 10 condition reports that each aggregated a number of individual problem identification or condition reports that had 1 or more common attributes. The issues addressed in these aggregate condition reports included:

- Enhancements to the corrective action program,
- Post-maintenance testing program weaknesses,
- Adverse human performance trend,
- Personnel assigned work without proper qualifications,
- Maintenance department human performance trends,
- Lack of strong industrial safety culture, and
- Training deficiencies in engineering.

The inspectors noted a recurring causal factor cited in a number of the aggregate condition reports. Six of the condition reports referred, in one manner or another, to lack of ownership, not accepting responsibility, low accountability, and low management support or involvement as underlying existing problems. The inspectors interviewed members of the licensee=s staff, including the training manager, who stated that this had been recognized by the licensee as a site-wide issue. The training manager stated that part of the problem arose from regional cultural behavior, which was to be non-confrontational. He also described the licensee=s attempts to address the problem, which included an application of an AAccountability Model.@ The model summarized a process of formulating standards and expectations, providing training, monitoring performance, and providing feedback. The licensee attempted a Atrickle-down@ approach that provided the initial training to the upper managers. The licensee expected that the upper managers would propagate the training throughout their respective organizations. The licensee stated that this approach did not achieve the desired result and that a new approach was being formulated. While this common problem did not result in risk significant findings during this inspection, the licensee clearly recognized the impact of this issue on its problem identification and resolution effectiveness. The inspectors agreed.

4OA4 Other

a. Inspection Scope

The inspectors interviewed 12 members of the licensee staff representing a cross-section of the licensee=s organization. Part of these interviews focused on the interviewee=s understanding of the corrective actions and employee concerns programs. Additionally, the inspectors assessed the experience of the system engineers who had oversight of the systems inspected.

b. Observations and Findings

Employee Concerns Program

The employee concerns program manager reported that the number of reported concerns had decreased from 25 in 1998 to 8 through July 1999. None of the concerns reported to the employee concerns program related to plant equipment or operational issues. There was no indication of generic issues from the concerns reported to date. Surveys conducted over the last year generated a 60 percent response that indicated over 90 percent satisfaction with the employee concerns program. The inspectors interviewed four individuals from operations and engineering regarding the employee concerns program. All the interviewees stated that the program was working well and that they had no reservations about identifying safety concerns through all means available onsite.

System Engineers

In Condition Adverse to Quality 97-1542, the licensee identified system engineer high turnover rate as a contributing factor to failures to adequately address service water system erosion, corrosion, and reliability problems.

During interviews, the inspectors identified that there continued to be a high turnover rate among system engineers and that those interviewed tended to have less than a year of oversight responsibility for their respective systems. There was no structure to the turnover of system information, but a strong reliance on the availability of the predecessor. For example, the individual had been the RHR system engineer for only 6 months. He did not turnover the responsibility with the former RHR system engineer but rather obtained the status of the system from the backup system engineer. The new system engineer did not review the problem identification report/corrective action program history for the RHR system beyond action items that were still open. Most of the system engineers interviewed had not researched the problem identification report/corrective action program history of their respective systems. Further, there was no expectation that system engineers routinely review problem identification report/corrective action program/nuclear action item tracking data for adverse trends. A few indicated that they sometimes trended specific system operating parameters via the plant monitoring computer.

Threshold and Prioritization of Problems

The primary method used for the identification and prioritization of problems was documented in Administrative Procedure 0.5, APProblem Identification and Resolution,@ Revision 19C2. This procedure prescribed the method for processing problem identification reports for the identification, documentation, notification, evaluation, correcting condition events, activities, and concerns that had the potential of adversely affecting safe and reliable operation of the Cooper Nuclear Station. The licensee=s corrective action program provided a multiple step process in which a screening of all problem identification reports. The daily screening meeting provided evaluations and recommendations to the condition review group, which determined the priority of the task (whether a significant condition report or resolve condition report needed to be initiated), and the responsible organization assigned to complete the assigned task. The inspectors found no instances of inadequate prioritization of work. In general, licensee

personnel utilized a conservatively low threshold for identifying and entering problems in the corrective action program.

The licensee=s trending program was delineated in Corrective Action Program Deskguide 6, AAdverse Trend Program,@ Revision 0. The licensee=s corrective action program group compiled and issued trended data to provide a measure of the corrective action program. The compiled results were provided monthly to plant management at a condition review group meeting. The inspectors observed that the reports contained data for the number of initiated problem identification, significant condition, and resolve condition reports, including the functional area and cause codes. Performance indicators for monitoring the corrective action program were established, which included the percentage of evaluations completed by the original due date, the number of extensions granted, and the percentage of significant condition report corrective actions, which were opened with an age that exceeded 1 year from initiation. On the basis of the sample reviewed, the corrective action program was generally implemented with an appropriate threshold for identifying and classifying adverse conditions.

Problem Identification and Resolution Conclusion

The corrective action program was generally implemented adequately across all cornerstones, with very low risk significance examples of untimely corrective actions. The licensee=s self-assessments were appropriately focused on substantive performance improvement areas. Licensee management identified improving ownership, accountability, and support as a site-wide improvement area and was developing improvement plans at the end of the inspection.

4OA5 Management Meetings

3. Exit Meeting Summary

The inspectors discussed the progress if the inspection on a daily basis and presented the inspection results to members of licensee management at the conclusion of the onsite inspection on September 10, 1999. The licensee=s representatives acknowledged the findings presented.

The inspectors asked the licensee staff and management whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

ATTACHMENT 1

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

Licensee

S. Blitchington, Supervisor ALARA
M. Boyce, Manager of System Engineering
D. Bremer, Operations Support Group
J. Burton, Performance Analysis Manager
P. Caudill, Senior Manager Technical Services
D. Cook, Training Manager
R. Church, System Engineer
K. Dorwick, Operations
L. Dugger, Manager of Engineering Support
J. Edom, Operations
R. Estrada, Corrective Action Program Supervisor
C. Fidler, Assistant Maintenance Manager
J. Flaherty, Assistant Design Engineering Manager
M. Gillan, Outage Manager
M. Hale, Senior Manager of Site Support
B. Houston, Operations Manager
L. Lockard, Supervisor Chemistry/Radiological
J. Long, Manager Nuclear Projects/Technical Services
M. Matheson, System Engineer
J. McDonald, Plant Manager
J. McMahan, Acting Work Control Manager
D. Montgomery, Omaha Public Power District
J. Peters, Licensing
B. Rash, Senior Manager of Engineering
R. Sessoms, Senior Manager Quality Assurance
G. Smith, Manager Nuclear Projects
S. Stiers, Manager Administrative Services
J. Sumpter, Licensing Supervisor
J. Swailes, Vice President
R. Wachowiak, Supervisor of Risk Management
C. Walgren, Senior Quality Assurance Auditor

NRC

M. Miller, Senior Resident Inspector
J. Pellet, Chief, Operations Branch
W. Sifre, Resident Inspector

INSPECTION PROCEDURES USED

IP 71152 Identification and resolution of Problems

ITEMS OPENED AND CLOSED

50-298/9903-01	NCV	Untimely corrective actions for the RHR system
50-298/9903-02	NCV	Failure to perform radiological survey

PROCEDURES

Procedure	Title	Revision
	Quality Assurance Program for Operation Policy Document	14a
AP 0.10	Operating Experience Program	5
AP 0.15	NPG Action Item Tracking	10
AP 0.40	Work Control Program	16
AP 0.5	Problem Identification and Resolution	19 C2
AP 0.27	Maintenance Rule Program	8
AP 0.27.2	Maintenance Rule Program Goal Setting	1
CAP Deskguide A	CAP Deskguide Administration	0
CAP Deskguide 1	PIR Processing	12
CAP Deskguide 2	SCR Evaluations	4
CAP Deskguide 3	Background Reviews and Trendcoding	12
CAP Deskguide 4	Conditions Requiring a PIR	0
CAP Deskguide 5	Problem Classification Guidance	1
CAP Deskguide 6	Adverse Trend Program	0
CAP Deskguide 7	Significant Condition Report and Resolve Condition Report Backend Reviews	3
CAP Deskguide 8	Significant Conditions Adverse to Quality Corrective Action Effectiveness Review	0
CAP Deskguide 9	Corrective Action Program Self Assessment Guideline	0

SOP 2.2.70	RHR Service Water Booster Pump System	41
SOP 2.2.69.2	RHR System Shutdown Operations	24
SOP 2.2.69.2	RHR System Shutdown Operations	32
6.EE.601	125V/250V Station and Diesel Fire Pump Battery 7 Day Check	9
6.EE.602	125V/250V Station and Diesel Fire Pump Battery 92 Day Check	7

Documentation Reviewed

Significant Condition Reports

97-1075	Compliance with Technical Specification 1.0.j@
98-0899	Maintenance rule evaluations
98-0214	Missed Technical Specification Surveillance Prior to Mode Change@
98-0268	Missed Technical Specification Surveillance of Average Power Range Monitoring System@
98-0285	Failure to SCRAM time test all RPS Relays@
98-0460	Missed Technical Specification Requirement of Testing Vacuum Breakers@
98-0611	Failure to Implement Radiological Controls for Drywall Head Removal@
98-0663	Foreign Material in the Service Water Side of Diesel Generator Lube Oil Cooler
98-1191	Training Deficiencies in Engineering
98-1235	Secondary Containment Isolation Valve not Verified Closed as Required by Improved Technical Specifications@
99-0465	Battery cell plate growth
99-0115	Maintenance Department Human Performance Trends
99-0148	Enhancements to the Corrective Action Program
99-0284	Post-maintenance Testing Program Weaknesses
99-0346	Failure of SRV Serial Number 387 to Meet TS $\pm$ 3% Auto Actuation Tolerance
99-0399	Failure to Adequately Test the Reactor Vessel Shroud Level Function@
99-0399	Failure to Adequately Test the Reactor Vessel Shroud Level Function

99-150 Lack of Strong Industrial Safety Culture

Significant Condition Adverse to Quality

98-0154 Seismic Classification of AZ@ Sump and Applicable Components

98-0358 Incorporation of pre-1991 design changes in emergency operating procedures

98-0368 SWBP AA@ Silting Problem

Resolve Condition reports

98-0767 RHR Heat Exchanger Waterbox Pitting

98-0837 Active Slim Layer in RHR Heat Exchangers - Accelerating Fouling

99-0311 RCIC-MO-132 valve failure to close

99-0452 Maintenance rule evaluations

99-0186 Configuration Control

99-0261 Lack of Fuse Sizing Criteria for MOVs

99-0262 Undersized Motor Starters on Valves RHR-27 A and B

99-0370 Adverse Human Performance Trend

Conditions Adverse to Quality Reports

97-0742 River Material in RHR Heat Exchanger B

97-0831 Failure to conduct RHR heat exchanger A performance test (Generic Letter 89-13)

97-0835 Fuel Bundles Moved to Racks Not Checked for Debris.

97-1133 Failure to Log F-Sump Integrator

97-1153 Failure to investigate as-found conditions for RHR heat exchanger B

97-1542 Incomplete/Inadequate Closure of Documents and Tasks Concerning

98-0092 Review of Radiological Job Packages@

98-0308 Multiple Errors Associated with Replacement of a Transverse Incore Probe@

98-0401 Response to EP Exercise Weakness regarding Contamination Controls

98-0431 Failure to Include Required Information in the Annual Radioactive Materials Release ReportA

Problem Identification Reports

4-02942	Battery cell plate growth
4-03002	Battery cell plate growth
4-03076	Battery cell plate growth
4-03172	Battery cell plate growth
4-03177	Battery cell plate growth
4-03361	Battery cell plate growth
4-03403	Battery cell plate growth

Licensee Event Reports

1998-005
1998-008
1998-013
1999-002
1999-005

Audit, Self Assessments

CAP Self-Assessment Report, May 1999
CNS Quality Assurance Audit Report AContinuous Improvement Program,@ Audit 99-09
CNS Quality Assurance Audit Report AContinuous Improvement Program,@ Audit 98-17
CNS Quality Assurance Audit Report AContinuous Improvement Program,@ Audit 98-10
Quality Assurance Scheduled Surveillance S408-9902, Corrective Action Program
30-Day Response to QA Audit 99-09, AContinuous Improvement Programs@
30-Day Response to QA Audit 98-17, AContinuous Improvement Programs@
30-Day Response to QA Audit 98-10, AContinuous Improvement Programs@
Response to QA Surveillance S408-9902, Corrective Action Program
Self-Assessment of the CNS Corrective Action Program, October 19, 1998
Final Site-Wide Assessment Report, December 7-18, 1998

Various

Erosion/Corrosion and Reliability of Service Water Components

CNS Business Plan, Initiative to Improve the Corrective Action Process, July 21, 1999

Performance Improvement Plan, Improve Effectiveness of the CNS Corrective Action Program, Revision G

USAR Volume II, Section IV-8.5.2, Residual Heat Removal, July 22, 1994

USAR Volume IV, Section X-8.2.5 Service Water Systems, May 25, 1999

CED 1998-0268, Addition of Throttling Capability for RHR-MOV-MO12A and -MO12B

Conceptual Design Document, Evaluation of Service Water Flow Through Valves SW-MO-89 A(B), Revision 0.

Contract 99-32, Service Water Booster System Valves@

PWG-01, Procedures Writer=s Guide, Revision 0

ATTACHMENT 2

INITIAL MATERIAL REQUESTED

Initial material requested for the 71152 inspection at Cooper Nuclear Station:

1. Index of all corrective action documents (e.g., CR's, CI-s, PIRs, etc.) for the last 12 months, including work requests, work orders, temporary modifications, calibration failures, and test failure reports.

2. All corrective action documents in response or related to plant transients or operating events, the equipment malfunctions that initiated them or that were previously written on the same equipment, for the last 12 months.

3. All corrective action documents related to external operating experience, especially vendor information not included in plant procedures and documents, for the last 12 months.

4. All major corrective action documents (i.e., those that subsume or integrate one or more smaller issues), for the last 12 months.

5. All corrective action documents associated with:
   1. Class 1E 125 Vdc System
   2. Automatic Depressurization System (ADS) & Safety Relief Valves (SRV)
   3. Residual Heat Removal (RHR) System
   4. Maintenance rule implementation, preventable functional failures, and baseline inspection results
   5. Operator workarounds.

6. All corrective action documents associated with non-escalated no response required or non-cited violations, for the last 2 years

7. All Maintenance Rule periodic evaluation reports for the past 12 months.

8. All quality assurance audits and self assessments for the last 12 months.

9. All corrective action program tracking and effectiveness internal reports or metrics, for the last 12 months.

10. Description of any informal systems, especially used by operations, for issues below the threshold of the formal corrective action system (e.g., CR's) and the content of those systems.

11. All procedures governing or applying to the corrective action program.