UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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December 3, 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. 50-298/99-11

Dear Mr. Swailes:

This refers to the inspection conducted on October 4 to 29, 1999, at the Cooper Nuclear Station facility. This was a Safety System Design and Performance Capability Inspection, that was performed in accordance with Inspection Procedure 71111.21, under the pilot plant study for the risk informed baseline inspection program. The primary objective of this inspection was to assess the adequacy of calculations, analyses, and other engineering activities used to support operability and reliability of the service water and 125 Vdc systems in the performance of the safety functions required by their design bases. The enclosed report presents the results of this inspection.

The inspection found that engineering activities supported the safe and reliable operation of the systems. No violations of NRC requirements were identified during the inspection.

One unresolved item was identified concerning the thrust ratings of motor-operated valves subject to elevated temperatures as a result of postulated high energy line breaks. This matter is discussed in Section 1R21.5b. of the enclosed inspection report.

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

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

Sincerely,

original signed by

Dr. Dale A. Powers, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

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

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NRC Inspection Report No.
50-298/99-11

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-298
License No.:	DPR 46
Report No.:	50-298/99-11
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	October 4 to 29, 1999
Team Leader:	C. Paulk, Senior Reactor Inspector Engineering and Maintenance Branch
Inspectors:	P. Goldberg, Reactor Inspector Engineering and Maintenance Branch W. McNeill, Reactor Inspector Engineering and Maintenance Branch M. Runyan, Senior Reactor Inspector Engineering and Maintenance Branch J. Whittemore, Senior Reactor Inspector Engineering and Maintenance Branch
Accompanying Personnel:	R. Quirk, Contractor Beckman and Associates
Approved By:	Dr. Dale A. Powers, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

Cooper Nuclear Station
NRC Inspection Report No. 50-298/99-11

The report includes the results of a team inspection of the service water and 125 Vdc systems.

Cornerstone: Mitigating Systems

- GREEN[1]. A leak from the reactor equipment cooling system was found to be the result of leaking tubes in a room cooler in the northeast quadrant of the secondary containment building. This was considered to be potentially significant because the reactor equipment cooling system is required to be capable of providing cooling for 30 days without makeup water. This issue was considered GREEN in the significance determination process since it did not represent an actual loss of safety function of a system, of a single train for more than the technical specification allowable outage time, or of a single train of non-technical specification equipment designated as risk-significant under 10 CFR 50.65 for more than 24 hours.

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[1]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 risk to safety, and which may require additional NRC inspection. 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 to shut down. The findings, considered in total with other inspection findings and performance indicators, will be used to determine overall plant performance.

REPORT DETAILS

1. REACTOR SAFETY

1R21 Safety System Design and Performance Capability

Introduction

Inspection of safety system design and performance verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected system to perform its design basis functions. As plants age, their design bases may be lost, such that, an important design feature may be altered or disabled. The plant risk assessment model is based on the capability of the as-built safety system to perform its intended safety function successfully. This inspectable area verifies aspects of the mitigating systems cornerstone for which there are no indicators to measure performance.

The objective of this safety system design and performance capability inspection was to assess the adequacy of calculations, analyses, other engineering documents, and maintenance practices that were used to support the performance of the service water and the 125 Vdc systems during normal, abnormal, and accident conditions. The inspection was performed by a team of inspectors that consisted of a team leader, four Region IV specialist inspectors, and a contractor. The systems, structures, and components examined during the inspection were selected by reviewing the licensee's probabilistic risk analysis model to determine the dominant systems, structures, and components, ranked by importance, and their potential contribution to dominant accident sequences and/or initiators. Acceptance criteria utilized by the NRC inspection team included the Cooper Nuclear Station technical specifications, applicable sections of the Updated Safety Analysis Report, Section XI of the ASME Code, industry initiatives implemented by the licensee, licensee procedures, and the design bases for the systems.

1R21.1 System Requirements

1. Inspection Scope

The team reviewed the following attributes for the service water and 125 Vdc systems: process medium (water, air, electrical signal), energy sources (electrical and air), control systems, operator actions, environmental requirements, equipment protection, and heat removal. These activities consisted of system walkdowns; reviews of normal, abnormal, and emergency operating procedures; and review of the Updated Safety Analysis Report, the technical specifications, design basis documents, and plant drawings. The purpose of these reviews was to verify that the service water and 125 Vdc systems' needs were met.

2. Observations and Findings

As a result of a continuously dropping water level in the reactor equipment cooling system surge tank, licensee operators periodically added water to the system to meet applicable design requirements. At the time of the inspection, there was an Updated Final Safety Analysis requirement that there shall be sufficient water in the closed loop reactor equipment cooling system surge tank to fulfill its safety requirement to supply water for 30 days following a loss-of-coolant accident. In June 1999, the licensee submitted a proposed license amendment request to the NRC to revise this requirement.

Normally, nonsafety-related demineralized water was used to fill the closed loop reactor equipment cooling system. The June 15, 1999, proposed license amendment changed this requirement to require that the reactor equipment cooling system fulfill its safety requirement for 7 days following a loss-of-coolant accident. The open loop service water system was to fulfill the safety function for the remaining duration of the accident. The maximum reactor equipment cooling system water leakage rate was based on the reactor equipment cooling system fulfilling its safety requirement for the 30-day period.

The proposed Updated Final Safety Analysis Report change would allow the maximum allowable reactor equipment cooling system leakage during normal power operation to increase, but the reactor equipment cooling system surge tank would assure that the reactor equipment cooling system would fulfill its safety function for at least the first 7 days following a large break loss-of-coolant accident. The team noted that the use of the service water system would result in silty, muddy water flowing through the reactor equipment cooling system coolers, which would increase the fouling of the heat exchangers, thereby reducing their capability to remove heat.

Licensee personnel identified, on September 16, 1999, leaks in the safety-related room cooler located in the northeast quadrant of the secondary containment building, which contained two residual heat removal pumps. Under analyzed conditions, loss of the room cooler could result in the eventual loss of the pumps because of overheating. This identified condition caused the licensee's engineers to consider the reactor equipment cooling system to be degraded.

Licensee personnel removed the coils, which were 25 years old, and replaced them with new coils. Three of the tubes at the U-bends were found by licensee personnel to have through-wall leakage. The team observed that there was a corrosion layer in the U-bends. The team determined that there were two additional safety-related room coolers that were 25 years old and one that was 10 years old. Licensee personnel stated that reactor equipment cooling system tube side cleaning and inspection were not done, nor were they required, on these safety-related room coolers. The team concluded that this lack of cleaning could lead to a common mode failure as a result of corrosion build-up. (At the time of the inspection, samples of the tubes were sent out to an independent laboratory to determine the type of corrosion.)

The team reviewed Problem Identification Report 4-04166, in which licensee personnel addressed the quadrant room cooler tube failure problem. Other problem identification reports (4-04177, 4-04178, 4-04179, and 4-04181) were initiated by licensee personnel to address inspection of the other three safety-related room coolers. During the review of Maintenance Work Request 99-2764, the team noted that licensee personnel inspected the room cooler in the northwest quadrant of the secondary containment building. Licensee personnel inspected the coils of the room cooler and determined that there were no leak paths and no indications of any future leak paths. In addition, licensee personnel scheduled inspections for the remaining two coolers.

The team determined that the leaking room cooler affected the reactor equipment cooling water system, a safety-related system; and if no action was taken, the condition could worsen and the reactor equipment cooling water system might not be capable of fulfilling its safety function (i.e., provide cooling under accident conditions for a 30-day period).

Since the condition could worsen without any action, the team used the significance determination process for assessment. Using the plant system degraded conditions worksheet for the degraded cornerstone Mitigation Systems, the team found that the answer to the first question (AIs this issue a design or qualification deficiency that does NOT affect operability per GL 91-18 (rev 1)?@) to be YES. Therefore, the team determined that this deficiency screened out as GREEN.

1R21.2 System Condition and Capability

a. Inspection Scope

The team reviewed the following attributes for the service water and 125 Vdc systems: the installed configurations, maintenance associated with age degradation and service-related wear, system operation, tested parameters, environmental and seismic qualification, procurement, and operating experience. These activities consisted of system walkdowns; reviews of normal, abnormal, and emergency operating procedures; review of the Updated Safety Analysis Report, the technical specifications, design basis documents, and plant drawings; review of test results; and review of the environmental and seismic qualification programs.

2. Observations and Findings

There were no significant findings identified during this inspection.

1R21.3 Identification and Resolution of Problems

1. Inspection Scope

The team reviewed a sample of problems identified by the licensee in the corrective action program to evaluate the effectiveness of corrective actions. Inspection Procedure 71152, AIdentification and Resolution of Problems,@ was used as guidance.

2. Observations and Findings

There were no significant findings identified during this inspection.

1R21.4 System Walkdowns

1. Inspection Scope

The team performed walkdowns of the service water, reactor equipment cooling, and 125 Vdc systems. The walkdowns focused on the installation and configuration of piping, components, and instruments; the placement of protective barriers and systems; the susceptibility to flooding, fire, or other environmental concerns; the physical separation; the provisions for seismic concerns; and accessibility for operator action.

2. Observations and Findings

There were no significant findings identified during this inspection.

1R21.5 Design Review

1. Inspection Scope

The team reviewed the designs to verify that the systems would function as required under accident conditions. The team reviewed design assumptions, calculations, boundary conditions, and models. These reviews were performed to determine whether the design bases of the systems were met by the installed and tested configurations.

2. Observations and Findings

The team noted that, because of various modifications, recalculations, and changes to assumed river water temperatures, the maximum expected post-loss-of-coolant-accident ambient temperatures in the reactor building had increased from an approximate original specification of 60EC [140EF] to 68EC [155EF] in most compartments. The team verified that environmental qualification records had been satisfactorily updated to reflect the higher temperatures.

However, when questioned, the licensee=s representatives were unable to confirm that motor-operated valve motor-torque outputs had been properly derated for the revised ambient conditions. Limitorque**7**, the motor-operated valve vendor, had addressed the phenomenon of ac motor torque loss at elevated temperatures (above 40EC [104EF]) in its Technical Update 93-03, issued in September 1993. In response to the team=s questioning, a licensee engineer issued Problem Identification Report 4-04994 on October 29, 1999. The initial engineering review indicated that the temperatures assumed in motor-operated valve calculations were inconsistent with the environmental profiles and were nonconservative in most cases. Because of the small magnitude of torque loss over this range of temperatures and the existing available torque margins, the licensee=s engineers documented within the problem identification report that an operability impact was not anticipated. The team agreed that the issue did not appear to represent an immediate operability concern.

The team asked an additional question concerning motor-operated valve motor torque derating for the environmental conditions associated with high energy line breaks. This concern pertained only to motor-operated valves that were assumed, within the design basis, to actively operate in an environment caused by a high energy line break. The licensee=s representative was unable to state whether motor-operated valves had been adequately reviewed for motor derating under high energy line break conditions and indicated that this question would be added to the scope of Problem Identification Report 4-04994. The team believed that, if a motor-operated valve was assumed to operate in an environment caused by a high energy line break, but had not been evaluated under those conditions, an operability impact may exist.

Using the Plant System Degraded Conditions worksheet, the team determined that the mitigation cornerstone was the only cornerstone that was potentially degraded. If motor-operated valves exposed to high energy line breaks were not properly analyzed or designed for these conditions, they could fail to operate as assumed in the accident analysis. Depending on the valve affected, and the timing and duration of the inoperable condition, the functionality of certain mitigation systems could be compromised. The screening questions cannot be answered until additional information is received from the licensee. Consequently, this item was considered unresolved (50-298/9911-01) pending completion of the evaluation by the licensee=s engineers and subsequent review by the NRC.

4 OTHER ACTIVITIES

4OA4 Other

4OA4.1 (Closed) Inspection Followup Item 50-298/9610-01: additional justification for the valve factor assumption for Valve Groups 2, 3, and 8.

This item was initially opened to track additional efforts by the licensee to further justify the valve factor assumptions for Valve Groups 2, 3, and 8. The licensee performed additional testing during the 1997 refueling outage and validated or revised, as necessary, the valve factors assigned to Valve Groups 2, 3, and 8. No significant findings were identified.

4OA4.2 (Closed) Unresolved Item 50-298/9624-08: anticipated transient without scram emergency operating procedure issues.

During a safety system functional inspection conducted October 7, 1996, through February 19, 1997, NRC inspectors identified three concerns with Procedure 5.8, "Emergency Operating Procedures," Revision 8. Two of these issues were subsequently reviewed and closed in NRC Inspection Report 50-298/98-15. The third issue, which remained open, was related to the displacement of boron after the standby liquid control system was used to shutdown the reactor, and the shutdown cooling mode of the residual heat removal system was used to bring the reactor to a cold shutdown condition.

The Office of Nuclear Reactor Regulation (NRR) provided, in a memorandum dated January 14, 1998, an evaluation of industry-wide strategy for handling an anticipated transient without scram. In the memorandum, NRR stated that the Boiling Water Reactor Owners' Group emergency operating procedure committee had been contacted and the NRC would continue to monitor the deliberation of this issue.

The team considered the issue of inadequate mixing of boron when placing the residual heat removal system into the shutdown cooling mode after using the standby liquid control system to shutdown the reactor to be a degraded condition. The degraded condition was the result of potential dilution of boron in the reactor coolant system when the residual heat removal system is placed in service. The team evaluated this condition with the significance determination process.

The issue of mixing related to boron dilution during initial operation in the shutdown cooling mode has not been evaluated. The standby liquid control system was considered to be the system with the degraded condition. This condition has existed for greater than 30 days. The team determined that the mitigation systems cornerstone was the affected cornerstone on the basis of the effect on reactivity control.

The team then considered the Phase 1 screening process for the mitigation systems cornerstone. While the team considered the issue to be a design deficiency that did not affect operability, the licensee had not performed an operability evaluation in accordance with the Generic Letter 91-18 program. Therefore, the team answered the first question as "NO." The issue did not represent either an actual loss-of-safety function of a system, an actual loss-of-safety function of a single train for a time that exceeded the allowable outage time for a technical specification, or an actual loss-of-safety function of a single train of nontechnical specification-related equipment considered as risk-significant by the Maintenance Rule program for greater than 24 hours. Therefore, the team answered the second, third, and fourth questions as "NO." As such, this issue had low safety significance.

4OA4.3 (Closed) Unresolved Item 50-298/9624-11: potential bypass of containment through the control rod drive system.

During a safety system functional inspection conducted October 7, 1996, through February 19, 1997, NRC inspectors reviewed a modification to install two check valves in the potential leakage path. This modification was developed in response to the licensee's evaluation of General Electric Potential Reportable Condition 89-15, "Control Rod Drive System Leakage During a LOCA."

The inspectors were concerned that the licensee used the plant-specific source term in lieu of the source term provided by Regulatory Guide 1.3, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss of Coolant Accident for Boiling Water Reactors." The use of the plant-specific source term resulted in a lower calculated exposure. By memorandum dated March 3, 1997, Region IV requested NRR to evaluate the licensee's treatment of potential control rod drive bypass leakage. NRR provided its response by memorandum dated May 24, 1999.

In the May 24, 1999, memorandum, NRR determined that the licensee's use of the plant-specific source term was acceptable. The licensee determined that the implementation of the modification would improve plant and public safety by preventing possible control rod drive bypass leakage past secondary containment boundaries. On the basis that NRR found the use of the plant-specific source term to be acceptable, the team found that there was no violation of regulatory requirements.

4OA4.4 <u>(Closed) Violation 50-298/9624-14</u>: inadequate or no safety evaluations. The team verified the corrective actions described in the licensee's response letter, dated July 24, 1997, to be reasonable and complete. No similar problems were identified.

4OA4.5 <u>(Closed) Licensee Event Report 50-298/97001-00</u>: six containment penetrations susceptible to thermally induced over pressurization. This licensee event report discussed a minor issue.

4OA4.6 <u>(Closed) Inspection Followup Item 50-298/97201-07</u>: duration of the residual heat removal system in the suppression pool cooling mode.

During an architect engineering inspection, the inspectors questioned the amount of time that the licensee operated the residual heat removal system in the suppression pool cooling mode. The issue was referred to NRR for review.

NRR has determined that infrequent operation of the residual heat removal system in the suppression pool cooling mode is included in the design basis. This includes both periodic short-term operation and long-term post-loss-of-coolant-accident operation. Additionally, NRR concluded that the system would be considered operable when in the suppression pool cooling mode because the system was designed for automatic alignment to the low pressure core injection mode if the system was in the suppression pool cooling or test modes.

4OA4.7 <u>(Closed) Inspection Followup Item 50-298/97201-12</u>: inadequate consideration of instrument uncertainties.

During the architect engineering inspection, the inspectors identified potential deficiencies in the licensee's accounting for instrument uncertainties when determining compliance with technical specifications. The licensee acknowledged, in Problem Identification Report 2-13343, dated October 20, 1997, that instrument uncertainties had not been taken into account when establishing the acceptance criteria for surveillance tests. In particular, instrument uncertainty had not been considered for Recorder FR-143 (residual heat removal system flow) and Indicator MI-TR-3020 (service water temperature).

The licensee evaluated this condition in Significant Condition Adverse to Quality 97-1407 and Condition Adverse to Quality 97-1453. The licensee concluded that no technical specification limits were compromised as a result of not considering instrument uncertainties.

The team considered this to constitute a potentially degraded condition and implemented the significance determination process. The team found that the condition had existed for greater than 30 days and affected the residual heat removal and service water systems. The team determined that the affected cornerstone was Mitigation Systems. Since this evaluation determined that the design deficiency did not affect the operability of technical specification instruments, the team answered the first question as "YES." Therefore, this issue was of low safety significance.

4OA4.8 <u>(Closed) Inspection Followup Item 50-298/97201-14</u>: condensate storage tank technical specification requirements.

During the architect engineering inspection, the inspectors identified a potential discrepancy between the Updated Safety Analysis Report and the technical specifications. Updated Safety Analysis Report, Section 4.4, stated that refueling operation could be conducted with the suppression pool drained, provided that an operable core spray or low pressure core injection subsystem was aligned to take suction on Condensate Storage Tank 1A, which must contain at least 567,812 liters [150,000 gallons] of water. Technical Specification 3.5.F.5.c required that 870,645 liters [230,000 gallons] be available in the condensate storage tank with one control rod drive housing open while the suppression pool chamber is completely drained. Technical Specification 3.10.F required that 567,812 liters [150,000 gallons] be available in the condensate storage tank when the suppression pool chamber is completely drained. Technical Specification Bases 3.5 stated that, under the worst-case leak conditions, water inventory in the reactor, spent fuel pool, and condensate storage tank was required to provide 60 minutes of core cooling and sufficient water inventory to permit the water, which has drained from the vessel, to fill the torus to a level above the core spray and low pressure cooling injection suction strainers.

The team noted that the licensee had implemented the improved technical specifications since the architect engineering inspection was performed. The improved technical specifications do not have a specific counterpart to Technical Specification 3.5.F.5.c. Improved Technical Specification Surveillance Requirement 3.5.2.1.b allows only one required subsystem to be available during an outage.

Improved Technical Specification Surveillance Requirement 3.5.2.1.a then requires the suppression pool to be available as a source of water for the other required low pressure injection and spray subsystems. The improved technical specifications do not permit operations having the potential for draining the reactor vessel without additional sources of water. The team found that the licensee's current calculations and acceptance criteria support the improved technical specifications. Therefore, this issue is not safety significant.

4OA4.9 (Closed) Inspection Followup Item 50-298/97201-15: emergency core cooling system leakage into interfacing systems.

During the architect engineering inspection, the inspectors identified that Valves RHR-V-98 and -99 provide isolation between the residual heat removal system and the condensate storage tank, but have no inservice testing requirements. The team considered this condition to be a degraded condition and implemented the significance determination process.

The team found that the affected cornerstone was the Containment Barrier because of the potential bypass of containment. Subsequent to the architect engineering inspection, the licensee had an evaluation performed to qualitatively review various bypass paths. The team found the evaluation provided the licensee with confidence that the original licensing basis was acceptable. As a result of the evaluation, the licensee determined the containment system to be operable.

The team consulted with a senior reactor analyst in Region IV during the screening process because of the containment barrier cornerstone being affected. The senior reactor analyst informed the team that the screening process for the containment barrier cornerstone could include the acceptance of an operability determination to screen out an issue. With this information, the team screened this issue out as being of low safety significance on the basis that the design deficiency did not affect operability.

4OA4.10 (Closed) Inspection Followup Item 50-298/97201-17: passive failure of an emergency core cooling system pump seal.

During the architect engineering inspection, the inspectors raised a question related to a potential passive failure of an emergency core cooling system pump seal during the long-term cooling mode following a loss-of-coolant accident. The inspectors questioned how the long-term passive failure of a seal was addressed in the emergency core cooling system design and sump pump operating requirements. This issue was referred to NRR for evaluation.

The Probabilistic Safety Assessment Branch of NRR provided the results of the review to Region IV by a memorandum dated March 30, 1999. In the memorandum, NRR concluded that the licensee was not required to assess this single passive failure in the radiological analyses. On the basis of this conclusion, no degraded condition existed.

4OA4.11 (Closed) Unresolved Item 50-298/97201-18: reactor building sump pump seismic qualification.

During the architect engineering inspection, the inspectors questioned the qualification of the reactor building floor drain sump pumps because there were no identified licensee requirements to maintain the pumps in a qualified configuration. The team found that, while there were no specific requirements for the licensee to maintain the qualification, the licensee has maintained the pumps in a qualified configuration. The team did not identify any violation of regulatory requirements. Therefore, there was no degraded condition.

4OA4.12 (Closed) Inspection Followup Item 50-298/97201-25: reactor equipment cooling system design.

During the architect engineering inspection, the inspectors questioned the design of the reactor equipment cooling system. The inspectors identified an apparent discrepancy between the Updated Safety Analysis Report and the safety evaluation report issued by the Atomic Energy Commission on February 14, 1973.

This issue was referred to NRR for review. On September 3, 1999, the NRR project manager for the Cooper Nuclear Station was provided the results of the review in an electronic mail message from Mr. J. Tatum. The message stated that the Updated Safety Analysis Report was not clear about this matter. Therefore, the Updated Safety Analysis Report should be revised to eliminate the confusion.

The team found that there was no violation of regulatory requirements, nor a degraded condition.

4OA4.13 (Closed) Inspection Followup Item 50-298/97201-28: electrical separation of reactor equipment system cables.

During the architect engineering inspection, the inspectors questioned the interconnection between the reactor equipment cooling system and the service water system to control the interconnecting valves from intertie switches in the control room. The inspectors noted that the Division II intertie switch operated Valves REC-MOV-714, REC-MOV-698, SW-MOV-887, and SW-MOV-889. With the exception of Valve REC-MOV-698, which was powered from a Division I motor control center, the valves were powered from a Division II motor control center.

The inspectors had noted that the design criteria document provided criteria for the design of cable and wiring systems that were protected from single failures. However, they also noted that the document did not specify design methods for associated cables, nor did it provide any guidelines defining acceptable methods for separation of circuits.

The team found that the licensee was not committed to Regulatory Guide 1.75, APhysical Independence of Electric Systems.@ Therefore, there was no regulatory requirement for the licensee to address the concerns raised during the architect engineering inspection. The team determined that the licensee=s design met the regulatory requirements and no degraded condition existed.

4OA5 Management Meetings

.1 Exit Meeting Summary

The team leader presented the inspection results to members of licensee management during a telephonic inspection exit on November 30, 1999. The licensee representatives acknowledged the findings presented.

The team leader asked the licensee representatives whether any materials examined during the inspection should be considered proprietary. No proprietary information was identified.

<u>ATTACHMENT</u>

SUPPLEMENTAL INFORMATION

PARTIAL LIST OF PERSONS CONTACTED

<u>Licensee</u>

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L. Dugger, Manager, Engineering Support
S. Freborg, ESD Valve Engineer
W. Frewin, Design Engineering Manager
D. Madsen, Senior Licensing Engineer
J. Peters, Licensing
B. Rash, Senior Manager, Engineering
J. Swailes, Vice President - Nuclear

<u>NRC</u>

J. Clark, Senior Resident Inspector
M. Hay, Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

<u>Opened</u>

50-298/9911-01	URI	Evaluation of effects on motor-operated valves subject to high energy line breaks (Section 1R21.5).

<u>Closed</u>

50-298/9610-01	IFI	Additional justification for the valve factor assumption for Valve Groups 2, 3, and 8 (Section 4OA4.1).
50-298/9624-08	URI	Anticipated transient without scram emergency operating procedure issues (Section 4OA4.2).
50-298/9624-11	URI	Potential bypass of containment through the control rod drive system (Section 4OA4.3).
50-298/9624-14	VIO	Inadequate or no safety evaluations (Section 4OA4.4).
50-298/97001-00	LER	Six containment penetrations susceptible to thermally induced over pressurization (Section 4OA4.5).
50-298/97201-07	IFI	Duration of the residual heat removal system in the suppression pool cooling mode (Section 4OA4.6).

50-298/97201-12	IFI	Inadequate consideration of instrument uncertainties (Section 4OA2.7).
50-298/97201-14	IFI	Condensate storage tank technical specification requirements (Section 4OA4.8).
50-298/97201-15	IFI	Emergency core cooling system leakage into interfacing systems (Section 4OA4.9).
50-298/97201-17	IFI	Passive failure of an emergency core cooling system pump seal (Section 4OA4.10).
50-298/97201-18	URI	Reactor building sump pump seismic qualification (Section 4OA4.11).
50-298/97201-25	IFI	Reactor equipment cooling system design (Section 4OA4.12).
50-298/97201-28	IFI	Electrical separation of reactor equipment system cables (Section 4OA4.13).

LIST OF BASELINE PROCEDURES PERFORMED

Inspection Procedure 71111-21 Safety System Design and Performance Capability

Inspection Procedure 71152 Identification and Resolution of Problems

LIST OF DOCUMENTS REVIEWED

Procedures

PROCEDURE NUMBER	DESCRIPTION	REVISION
	Service water and residual heat removal service water booster system design criteria document	2
0.10	Operating Experience Procedure	5
0.11	10 CFR Part 21 Evaluations	3
0.20	Environmental Qualification of Electrical Equipment	5
0.26	Surveillance Program	36
0.41	Seismic Housekeeping	0
0.8	Safety Assessments and Unreviewed Safety Question	6

PROCEDURE NUMBER	DESCRIPTION	REVISION
	Determinations	
1.4	Procurement Procedure	18
2.1.11	Station Operators Tour	89
2.2.3	Circulating Water System	51
2.2.42	Service Water Pump Room High Temperature	4
2.2.71	Service Water System	45
2.4.8.3.1	Service Water System Casualties	11
3.12.1	Equipment Qualification Program Implementation	6
3.12.2	Equipment Qualification Data Package	11
3.12.3	Equipment Qualification File Control	8
3.12.5	Age Related Degradation Evaluation For Equipment Qualification	8
3.12.7	Control of Master Equipment List (MEL)	4
3.15	Procurement Document Review	2
3.4.7	Design Calculations	15
3.4	Configuration Change Control	27 C1
3.26	Procedure, Instrument Setpoint Control and Meter Banding Control	12
3.26.3	Instrument Setpoint and Channel Error Calculation Methodology	4
4.12	Seismic Instrumentation	10 C1
5.1.4	Low River Level	4
5.2.3	Loss of All Service Water	13
5.2.5.1	Loss of All Site Power Station Blackout	12
5.3.10	Control Building Basement Flooding	14

PROCEDURE NUMBER	DESCRIPTION	REVISION
6.LOG.601	Daily Surveillance Log - Modes 1, 2, and 3	16
6.EE.609	125V/250V Station Battery Intercell Connection Testing	6
7.0.2	Preventive Maintenance Program	19
7.0.15	Station Painting Procedure	6 C1
7.2.42	Heat Exchanger Cleaning	12
7.3.31.3	125V/250V Battery Terminal Cleaning and Torquing	2
7.3.50.3	SW 89A/B Minimum Flow Adjustment	3
8.7.1.8	Biomonitoring	1
14.3.9	Seismic Monitoring System Testing	6.1

Problem Identification Reports

1-00091	2-16562	2-29695	4-03688	4-04964	98-3051
1-14242	2-19007	2-30178	4-03715	4-04992	98-3119
1-16180	2-19377	2-31315	4-03830	4-04993	98-3125
1-17274	2-19977	2-31949	4-03990	4-04994	99-0665
1-18238	2-22194	3-20476	4-04166	94-0089	99-1027
1-20798	2-22681	3-20780	4-04177	97-0567	99-1360
2-00505	2-23058	3-50334	4-04178	97-0989	99-1911
2-03715	2-23303	3-50953	4-04179	97-1475	99-1992
2-04147	2-23304	3-50957	4-04181	97-0066	99-2006
2-04148	2-23517	4-00273	4-04325	97-1407	99-2022
2-08454	2-23910	4-00579	4-04613	97-1453	
2-10012	2-24633	4-00837	4-04643	98-0119	
2-12005	2-27614	4-00960	4-04644	98-1358	
2-13061	2-28041	4-01629	4-04685	98-2660	
2-13343	2-28382	4-02196	4-04955	98-2736	
2-14685	2-28823	4-02386	4-04957	98-2926	

Drawings

DRAWING NUMBER	DESCRIPTION	REVISION

DRAWING NUMBER	DESCRIPTION	REVISION
2006	Isometric Key Circulating Screen Wash & Service Water System, Sheets 1, 2, 3, and 5	N01, 0, N05, and N04
2006	Flow Diagram Circulating, Screen Wash & Service Water Systems, Sheets 1, 2, 3, 4, and 5	N42, N13, N37, N35, and N17
2007	Turbine Building Closed Cooling Water System	N56
2036	Isometric Key Reactor Building Service Water System	N05
2036	Reactor Building Service Water System Flow Diagram, Sheet 1	N68
2036	Reactor Building Service Water System Flow Diagram, Sheet 2	N07
2077	Diesel Generator Building Service Water, Starting Air, Fuel Oil, Sumo System, and Roof Drains Flow Diagram	N47
2851-1	18", Service Water-1, Class IV P, Class 1S Seismic, Control Building	N10
2851-2	18", Service Water-1, Class IV P, Class 1S Seismic, Control Building	N09
2851-6	18", Service Water-1, Class IV P, Class 1S Seismic, Control Building	9
2851-7	18", Service Water-1, Class IV P, Class 1S Seismic, Control Building	N08
2852-1	24" SW-2 to Heat Exchangers 1-A & 1-B Class II-P Turbine Building	N03
2852-3	Pump Room SW-2 Discharge Intake Structure - Class IV	N12
2852-9	SW-2 Service Water	N08
2852-18	1-2" SW-2 Class IV-P, Class IS Seismic - Control Building	N10
2852-19	14" SW-2 Class IV-P, Class IS Seismic - Control Building	N08

DRAWING NUMBER	DESCRIPTION	REVISION
2852-24	SW-2 Service Water Diesel Generator Building	N02
3002	Auxiliary One Line Diagram MCC Z, SWGR BUS 1A, 1B, 1E, & Critical Switchgear Bus 1F, 1G, Sheet 1	N32
3006	Auxiliary One Line Diagram Starter Racks LZ and TZ MCC=s K, L, LX, RA, RX, S, T, TX, X Sheet 5	N65
3006	Cooper Nuclear Station Auxiliary One Line Diagram Starter Racks LZ and TZ MCC=s K, L, LX, RA, RX, S, T, TX, X, Sheet 6	N65
3007	Auxiliary One Line Diagram MCC=s E, Q, R, RB and Y Sheet 6	N70
3019	4160V Switchgear Elementary Diagram (Includes Second Level Undervoltage Protection) Sheet 3	N26
3022	4160v Switchgear Elementary Diagram (Includes SW Pump) Sheet 6	N22
3023	4160v Switchgear Elementary Diagram (Includes RHR SW Booster Pump) Sheet 7	N15
3037	Control Elementary Diagrams Sheet No. 6 (Includes SW-MO-650, -651)	N26
3040	Control Elementary Diagrams (Includes SW-MO-36MV) Sheet 9	N22
3058	Cooper Nuclear Station DC One Line Diagram	N41
3059	DC Panel Schedules	N31
3067	Control Elementary Diagram Sheet 19 (Includes SW-M0-37MV, Control Building Basement Flood Water Level LS-751, Service Water REC HX TCV 451A & 451B)	N30
3068	Control Elementary Diagram Sheet 20 (Includes SW Pump Bearing Injection)	N14
3070	Electrical Symbol List	03
3072	Cooper Nuclear Station Control Elementary Diagram	N04
	Cooper Nuclear Station Auxiliary One Line Diagram MCC CA, CB	

DRAWING NUMBER	DESCRIPTION	REVISION
3401	MR, OG1 & OG2	26
791261	Relay Logic Circuit A	N15
791261	Relay Logic Circuit B	N15
E150	Relay Settings for Bus 1F Sheet 7	N22
E150	Relay Settings for Bus 1G Sheet 9	N25
G5-262-743	Emergency Diesel Generator #1 Electrical Schematic Sheet 1	N15
G5-262-743	Emergency Diesel Generator #2 Electrical Schematic Sheet 10	N10
G5-262-743	Emergency Diesel Generator #2 Electrical Schematic Sheet 10A	N01
KSV-47-8	Diesel Generator Cooling Water Schematic	N18
KSV-48-5	Starting Air Schematic Sheet 1	N11
M-1	Emergency Diesel Generator Flow Diagram	N01
X-2851-209	SW-1 Service Water	N06
X-2851-212	SW-1 Service Water	N01
X-2852-200	SW-2 Service Water	N02
X-2852-202	SW-2 Service Water	N01
X-2852-211	SW-2 Service Water	N02
X-2852-218	SW-2 Service Water	N02
X-2852-230	SW-2 Service Water	N03
X-2852-234	SW-2 Service Water	N01
X-2852-235	SW-2 Service Water	N01
X-2852-236	SW-2 Service Water	N01

DRAWING NUMBER	DESCRIPTION	REVISION
X-2852-238	SW-2 Service (River) Water	N01
X-2852-239	SW-2 Service (River) Water	N01
X-2870-355	RWD-1-River Well Discharge	N08

Calculations

CALCULATION NUMBER	TITLE	REVISION
46	Equipment Qualification Data Package Environmental Conditions	5
220	Equipment Qualification Data Package Temperature Switch	7
223	Equipment Qualification Data Package Temperature Elements	7
86-105D	Critical DC Bus Coordination Study	7
91-239	Review of APA DG Jacket Water, Lube Oil and Intercooler Heat Exchanger Calculations	0
92-050AE	SW-PS-364A/B Setpoint Calculation	1
98-002	REC Surge Tank DBA LOCA Volume Requirement and Allowable REC System Uncontrolled Inventory Loss	1
99-073	DGLO/DGJW Heat Exchanger Evaluations at off Design Conditions	0
NEDC 87-131A	250 Vdc Division 1 Load and Voltage Study	8
NEDC 87-131B	250 Vdc Division 2 Load and Voltage Study	7
NEDC 87-131C	125 Vdc Division 1 Load and Voltage Study	8
NEDC 87-131D	125 Vdc Division 2 Load and Voltage Study	7
NEDC 87-132A	Plant AC Voltage Study	7

CALCULATION NUMBER	TITLE	REVISION
NEDC 89-149	Class IIN Main Steam Piping Analysis Problem MS-2	6
NEDC 90-388	NED Review of Hutch Calculation XNP031.0200.01, AFan Coil Unit Sizing Calculation@	1
NEDC 91-066	Flooding From Moderate Energy Line Breaks	1
NEDC 91-221	Service Water Pump Room Temperature After a Loss of Cooling	0
NEDC-91-232	Minimum Service Water Pump Room Temperature on Loss of Room Heating	0
NEDC 91-239	Review of APA DG Jacket Water, Lube Oil, and Intercooler Heat Exchanger Calculations	0
NEDC 92-093	CS Quad Temperature Rise	3
NEDC 92-0500	SW-PS-387(388) Setpoint Calculation	0
NEDC 93-050	RHR Quad Temperature with Hatches Removed	1
NEDC 93-093	Analysis of STP 93-062 Data (RHR Quad Heatup)	0
NEDC 93-184	Verification of Senior Engineering=s Calculation of the Thermal Performance of the RHR Heat Exchangers	0
NEDC 93-185	RHR Heat Exchangers= Thermal Performance During Power Generation Operations with an Increased Tube Plugging Margin	1
NEDC 94-021	REC-HX-A/B Maximum Allowable Accident Case Fouling	3
NEDC 94-034	Review of GE Nuclear Analyses GENE-673-020-0993 and GENE-637-045-1293 Supporting the Increase of the RHR Heat Exchangers Tube Plugging Margin	1
NEDC 96-029	Determination of SW Flows Under a LOOP/LOCA Scenario	0
NEDC 97-074	Evaluation of the Service Water System to Provide Direct Back-up Cooling to the REC System=s Critical Loops	1

CALCULATION NUMBER	TITLE	REVISION
NEDC 97-085	RHR Quad Heatup After Loss of Cooling with Two Pump Operation	2
NEDC 97-087	Acceptance Criteria for the Quad's Fan Coil Units (FCUs)	1

Maintenance Work Requests

93-0300	94-6656	95-2286	96-2063	97-0012	98-2933
94-5133	95-0742	95-3563	96-2144	97-2782	99-2764
94-5134	95-2061	96-1722	96-2145	97-2783	
94-5188	95-2062	96-2061	96-2250	98-0129	
94-6570	95-2064	96-2062	96-2251	98-0644	

Design Change Packages

NUMBER	DESCRIPTION	DATE/ REVISION
93-062	Hatch Plug Removal, Northwest and Southwest Quads	04/24/1993
94-377	Isolation of Non-Essential from Essential Loads	05/14/1998
95-036	SW Sequential Start Timer Setting Change	10/25/1996
CED 1999-0121	125Vdc Battery Cell Replacement (Temporary Change)	0

Engineering Judgements

NUMBER	DESCRIPTION	DATE
EJ 99-029	Engineering Judgement - Limiting Amperage Draw for Service Water Pump Motors	06/29/1999
EJ 99-030	Engineering Judgement - Limiting Amperage Draw for Service Water Pump Motors	06/29/1999

Miscellaneous Documents

DOCUMENT NUMBER	TITLE/DESCRIPTION	DATE/ REVISION
	Cooper Nuclear Station Generic Letter 89-10 MOV Program MOV Closeout Report	06/12/1996
	River Water Temperatures for Missouri River, June through September 1998 and 1999	
	Use of River Well Water for Various Pump Gland Supplies and Recommended Solution	03/10/1999
	Engineering Analysis of the Causes and Modality of Slamming in the RHR-SWBP Check Valves	08/19/1999
3-20780	SW Pump High Current Operabilty Evaluation	04/01/1999
6.EE.603	125V Battery Service Test of Battery A	10/24/1998
6.EE.603	125V Battery Service Test of Battery B	10/05/1998
6.EE.605	250V Battery Service Test of Battery A	10/25/1998
6.EE.605	250V Battery Service Test of Battery B	10/06/1998
6.EE.607	125V Station Battery Performance Discharge Test of Battery A	04/04/1997
6.EE.607	125V Station Battery Performance Discharge Test of Battery B	04/22/1997
6.EE.608	250V Station Battery Performance Discharge Test of Battery A	04/03/1997
6.EE.608	250V Station Battery Performance Discharge Test of Battery B	04/22/1997
CNSNO 36317	Technical Basis for the Use of Canady Battery Cells in Essential CNS Application	08/04/1999
DCD-3	Service Water Design Criteria Document	2
DCD-5	DC Electrical Distribution Design Criteria Document	3
DCD-35	Station Blackout Design Criteria Document	2