UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-8064

August 10, 2001

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

SUBJECT: COOPER NUCLEAR STATION - NRC INSPECTION REPORT 50-298/01-05

Dear Mr. Swailes:

On June 22, 2001, the NRC completed an inspection at your Cooper Nuclear Station. The enclosed report documents the inspection findings which were discussed on June 22, 2001, with Mr. J. Ranalli, Senior Manager of Engineering, and other members of your staff.

This inspection examined activities conducted under your license as they relate to safety and compliance with the Commission's rules and regulations and with the conditions of your license. Within these areas, the inspection consisted of selected examination of procedures and representative records, observations of activities, and interviews with personnel.

Based on the results of this inspection, the NRC has identified two findings that were evaluated under the risk significance determination process as having very low safety significance. One finding involved failure to establish undervoltage relay setpoints that were accurate and conservative with respect to technical specification requirements. The second finding involved failure to account for static head in determining the pressure switch setpoint for the Loop A residual heat removal keepfill system. The NRC has also determined that violations are associated with these two issues. These violations are being treated as noncited violations, consistent with Section VI.A.1 of the Enforcement Policy. The noncited violations are described in the subject inspection report. If you contest the violations or significance of the 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 U.S. 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, U.S. 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 and its enclosure will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/NRC/ADAMS/index.html (the Public Electronic Reading Room).

Sincerely,

/**RA**/

Charles S. Marschall, Chief
Engineering and Maintenance Branch
Division of Reactor Safety

Docket: 50-298
License: DPR-46

Enclosure:
NRC Inspection Report
50-298/01-05

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ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR 46
Report No.:	50-298/01-05
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	June 4-22, 2001
Team Leader:	C. J. Paulk, Senior Reactor Inspector
Inspectors:	R. W. Deese, Reactor Inspector
	L. E. Ellershaw, Senior Reactor Inspector
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	W. M. McNeill, Reactor Inspector
	M. F. Runyan, Senior Reactor Inspector
Accompanying Personnel:	Robert Quirk, Consultant Beckman and Associates, Inc.
Approved By:	Charles S. Marschall, Chief Engineering and Maintenance Branch Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000298/01-05, on 06/04-22/2001, Nebraska Public Power District, Cooper Nuclear Station, safety system design and performance capability and evaluation of changes, tests, or experiments.

The inspection was conducted by five regional inspectors and one contractor. The inspection identified two issues that were evaluated under the risk significance determination process as having very low safety significance (green). The significance of most findings is indicated by their color (Green, White, Yellow, Red) using IMC 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply are indicated by "No Color" or by the severity level of the applicable violation. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described at its Reactor Oversight Process website at http://www.nrc.gov/NRR/OVERSIGHT/index.html.

Cornerstone: Initiating Events, **Mitigating Systems**, **Barrier Integrity**

- Green. The measures established by the licensee for the translation of design requirements were not adequate to assure that the values used to establish the second level undervoltage relay setpoint were accurate and conservative with respect to the technical specifications. In addition, the measures for promptly identifying and correcting the adverse condition were not adequate as demonstrated by the length of time this condition has existed (since 1987). The failure to accurately translate design requirements was a violation of Criterion III of Appendix B to 10 CFR Part 50, and the untimely corrective actions was a violation of Criterion XVI of Appendix B to 10 CFR Part 50. This violation is noncited in accordance with Section VI.A of NRC's Enforcement Policy, and is in the licensee's corrective action program (Notification 10092429). (Section 1R21.5.b.1.)

  The finding was of very low safety significance because, although the calculated values were not conservative and were not consistent with the technical specification values, there were administrative procedures in place to prevent exceeding the correct analytical limit. Additionally, there was no actual loss of safety function.

- Green. The failure to properly account for the static head in Calculation NEDC 92-050AT, "CM-PS-270 Setpoint Calculation," Revision 0, resulted in the licensee adjusting Switch CM-PS-270, residual heat removal system, loop A keep fill system. The incorrect setting could have allowed a void in the keep fill line from being detected by the operators. This failure was a violation of Criterion III of Appendix B to 10 CFR Part 50. This violation is noncited in accordance with Section VI.A of NRC's Enforcement Policy, and is in the licensee's corrective action program (Notification 10089082). (Section 1R21.5.b.2.)

  The finding was of very low safety significance because there was no evidence that voids existed and, therefore, there was no actual loss of safety function.

Report Details

1. REACTOR SAFETY

Introduction

Inspections of safety system design and performance capability, and evaluations of changes, tests, or experiments were performed at Cooper Nuclear Station. These inspections were conducted to verify, respectively, that the initial design and subsequent modifications have preserved the design basis of selected system and related support systems, and that changes to the facility or procedures as described in the Updated Safety Analysis Report (USAR) and tests or experiments not described in the USAR are reviewed and documented in accordance with 10 CFR 50.59. Additionally, the inspection effort served to monitor the capability of the selected system to perform the design basis functions and to verify that safety issues pertinent to the changes are resolved. These inspectable areas verify aspects of the initiating events, mitigating systems, and barrier integrity cornerstones.

The probabilistic risk analysis for Cooper Nuclear Station is based on the capability of the as-built safety systems to perform their intended safety functions successfully. The area and scope of the inspection were predetermined by reviewing the licensee's probabilistic risk analysis to identify the risk-dominant systems, structures, and components, ranked by importance, and their potential contribution to dominant accident sequences and/or initiators. The primary review prompted a parallel review of support and interfacing systems, such as, electrical power.

The objective of this inspection was to assess the adequacy of calculations, analyses, other engineering documents, and engineering and operating practices that were used to support the performance of the residual heat removal (RHR) system. The inspection was performed by a team of inspectors that consisted of a team leader, inspectors, and a contractor. Acceptance criteria utilized by the NRC inspection team included those from Cooper Nuclear Station technical specifications, applicable sections of the Final Safety Analysis Report, applicable industry codes, and industry initiatives implemented by the licensee's programs.

1R02 Evaluations of Changes, Tests, or Experiments

a. Inspection Scope

The team reviewed a selected sample of eight safety evaluations to verify that the licensee had appropriately considered the conditions under which the licensee may make changes to the facility or procedures or conduct tests or experiments without prior NRC approval.

The team reviewed a selected sample of 11 safety evaluation screenings, in which the licensee determined that safety evaluations were not required, to ensure that the licensee's exclusion of a full evaluation was consistent with the requirements of 10 CFR 50.59, "Evaluations of Changes, Tests, or Experiments."

The team reviewed five problem identification reports initiated by the licensee that addressed problems or deficiencies associated with 10 CFR 50.59 requirements to ensure that appropriate corrective actions were being taken. The team also reviewed licensee self-assessments to ensure that problems or deficiencies were appropriately addressed.

b Findings

No findings of significance were identified.

1R21 Safety System Design and Performance Capability

.1 System Requirements

a. Inspection Scope

The team reviewed the following attributes for the residual heat removal (RHR) system: process medium (water and air), energy sources (electrical and air), control systems, and equipment protection. The team also reviewed applicable mechanical and electrical calculations. The team verified that procedural instructions to operators were consistent with operator actions required to meet, prevent, and/or mitigate design basis accidents.

To do this, the team reviewed abnormal and emergency operating procedures, and requirements and commitments identified in the USAR, the technical specifications (TSs), design basis documents, and plant drawings. The team reviewed alarm setpoints and verified that instrumentation and alarms were available to operators for making necessary decisions in coping with postulated accident conditions. In addition, the team verified that system alignments were consistent with design and licensing basis assumptions. The review also considered requirements and commitments identified in the USAR, the TSs, design basis documents, and plant drawings. The purpose of these reviews was to verify that the RHR system's needs were met.

b. Findings

No findings of significance were identified.

.2 System Condition and Capability

a. Inspection Scope

The team reviewed periodic testing procedures (listed in the attachment) and results to verify that the design requirements were demonstrated by the performance of tests. The team also verified the environmental qualification of a sample of system components for operation under design environmental conditions and assumed operating parameters (e.g., voltage, speed, and power).

The team also reviewed the system's operations by conducting system walkdowns; reviewing normal, abnormal, and emergency operating procedures; and reviewing the USAR, the technical specifications, design calculations, drawings, and procedures. In addition, the team reviewed the operations department list of active and closed standing orders and operator work-arounds to ensure no design assumptions were invalidated by past or current operator daily practices. The team critiqued the bases of each of the applicable standing orders and work-arounds.

Additionally, the team checked the licensee's operating experience review program to ensure applicable lessons learned dealing with similar events, systems, and components were incorporated into the applicable RHR system documentation and procedures.

b. Findings

No findings of significance were identified.

.3 Identification and Resolution of Problems (71152)

a. Inspection Scope

The team reviewed a sample of RHR system problems identified by the licensee in the corrective action program to evaluate the effectiveness of corrective actions related to design issues. The team also reviewed Administrative Procedure 0.5, "Corrective Action Effectiveness Reviews," Revision 2. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report. Inspection Procedure 71152, "Identification and Resolution of Problems," was used as guidance to perform this part of the inspection.

The team reviewed the actions the licensee has taken in response to industry-identified problems with the RHR system and support equipment.

b. Findings

No findings of significance were identified.

.4 System Walkdowns

a. Inspection Scope

The team performed selective field inspections of the RHR system. The purpose of these walkdowns was to assess the adequacy of materiel condition and installation configurations by focusing 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; physical separation; provisions for high energy line break; accessibility for operator action; and the conformance of the currently installed configuration of the systems with the design and licensing bases.

b. Findings

No findings of significance were identified.

.5 Design Review

a. Inspection Scope

Electrical, Instrumentation and Control

The team reviewed the electrical, and instrumentation and control aspects of the RHR system. The team reviewed electrical calculations for ac and dc power to selected emergency pumps and motor operated valves. In addition, the team performed a selective review of instrument setpoint and uncertainty calculations, as well as control circuits supporting initiation and control of the RHR system pumps and valves. The review included design assumptions, calculations, boundary conditions, and modifications.

The team also performed a single failure review of individual components to determine the potential effects of such failures on the capability of the system to perform its safety functions. Additionally, the team performed analyses to verify that design values were correct and appropriate, and translated into operational and maintenance procedures. Documentation reviewed included drawings, procedures, calculations, condition reports, and maintenance work orders identified in the attachment to this report, as well as the design bases document for the RHR system, the technical specifications, the Technical Requirements Manual (TRM), the USAR, operator training procedures, and risk analysis documents. The purpose of the reviews was to determine whether the design bases of the system were met by the installed and tested configurations.

Mechanical

The team reviewed the system's design to verify that the system would function as required under accident conditions. The review included design assumptions, calculations, boundary conditions, and modifications. The team also performed a single failure review of individual components to determine the potential effects of such failures on the capability of the system to perform its safety functions. Additionally, the team performed informal analyses in several areas to verify that design values were correct and appropriate. Documentation reviewed included drawings, procedures, calculations, safety evaluation reports, condition reports, and maintenance work orders identified in the attachment, as well as technical specifications, and the USAR. The team verified implementation of seismic requirements by reviewing engineering analyses and operating procedures governing the configuration of the components in the RHR system to ensure that their seismic qualification was maintained. The purpose of the reviews was to determine whether the design bases of the system were met by the installed and tested configurations.

b. Findings

Degraded Grid Protection Setpoint

The team identified a violation of 10 CFR Part 50, Appendix B, Criteria III and XVI for a nonconservative analysis of setpoints for second level undervoltage relays designed to respond to loss of offsite power conditions. The nonconservative settings did not result in an actual loss of safety function for the associated equipment, since the actual settings were adequate to insure proper operation. As a result, the issue had very low risk significance.

The Cooper engineering staff missed multiple opportunities to identify that the technical specification required setpoints for undervoltage relay setpoints were nonconservative. On July 17, 1991, the licensee issued Calculation NEDC 88-086B, "Setpoint Determination of Second Level Undervoltage Relays," Revision 4. Inspectors found that from the time of issuance of the calculation, it required a higher relay setpoint than the technical specification allowable value. This error could allow the technical specification to be met with the setpoint being less than the analytical value. Such a condition could result in plant operation at a voltage level that was insufficient to provide electrical power to safety-related equipment during design basis events. The licensee revised Calculation NEDC 88-086B three more times without discovering that the analytical value was greater than the technical specification allowable value. On October 18, 1996, Problem Identification Report (PIR) 2-05555 was issued in response to an industry question concerning the analytical limit and allowable value. For several reasons, including human performance problems, the PIR was closed, and other corrective action documents, including Condition Adverse to Quality Reports 97-0507 and 97-1452, and PIR 3-20476, were opened. All of these except PIR 3-20476 were closed, but the problem was not resolved.

In addition, the licensee used empirical data collected in 1987 to determine the analytical limit for the second level undervoltage relay setpoint. The licensee had collected the empirical data under conditions that did not reflect accident conditions. The nonconservative values were used in three sets of calculations of record that determined the second level undervoltage relay setpoint analytical limit. The calculations were: NEDC 88-086B, Revision 7, approved on December 7, 1993, NEDC 00-003, “Auxiliary Power System Low Flow and Voltage Analysis,” Revision 0, approved on December 5, 2000, and NEDC 00-111, “Auxiliary Power System AC Loads,” Revision 0.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that “[m]easures shall be established to assure that applicable regulatory requirements and the design basis . . . are correctly translated into specifications, . . . procedures, and instructions.”

Criterion XVI of Appendix B to 10 CFR Part 50 states, in part, that “[m]easures shall be established to assure that conditions adverse to quality . . . are promptly identified and corrected.

Contrary to the above, the measures established by the licensee for the translation of design requirements did not assure that the second level undervoltage relay setpoint was accurate with respect to the technical specifications. In addition, the measures for promptly identifying and correcting the adverse condition did not result in prompt correction.

The team concluded, therefore, that the failure to use correct design data and to promptly identify and correct the errors in Calculations NEDC 00-111, NEDC 88-086B and NEDC 00-003 was a violation of Criteria III and XVI of Appendix B to 10 CFR Part 50. The team evaluated this finding in accordance with NRC Appendix B of Inspection Manual Chapter 0610*, "Power Reactor Inspection Reports."

The team determined there was a credible impact on safety because the calculated values and the technical specification limits could have permitted plant operation with the second level undervoltage relay setpoint below the analytical limit. If a degraded grid condition existed with voltage below the analytical limit but above the second level undervoltage relay setpoint, vital plant equipment may not operate as required. Therefore the issue was a more than minor violation.

The team also concluded the issues affected the mitigating system cornerstone because vital equipment required to mitigate a design basis event may not operate or perform at the capacity assumed in the accident analyses. Therefore the significance determination process as described in NRC Inspection Manual Chapter 0609 was applied.

The team determined there was no actual loss of safety function because the actual second level undervoltage relay setpoint was set such that, even with total instrument uncertainty, the analytical limit would not be violated. The error in the calculations did have a credible impact on safety; however, since only the mitigating systems cornerstone was affected, and the actual setpoint was such that the analytical limit would not be violated, the finding is considered to be of very low safety significance (Green).

The team determined the failure to properly translate design basis electrical load information into Calculations NEDC 00-111 and NEDC 00-003, and the subsequent failure to promptly identify and promptly correct the issue, once identified, was a violation of Criteria III and XVI of Appendix B to 10 CFR Part 50. However, because of the very low safety significance, and because the licensee has included the item in their corrective action program (Notification 10092429), this violation is being treated as a noncited violation (50-298/0105 -01) in accordance with Section VI.A of the Enforcement Policy.

b.2 Residual Heat Removal Pressure Maintenance (Keep Fill) System

The team identified an error in Calculation NEDC 92-050AT, "CM-PS-270 Setpoint Calculation", Revision 0, that required the licensee to make an adjustment to a safety-related instrument.

The team determined that an error in Calculation NEDC 92-050AT, Revision 0, for Switch CM-PS-270, RHR Pump Discharge Line Low Pressure (Loop A), could have permitted voiding in the RHR Loop A without generating an alarm in the control room. Voiding could result in damage to RHR pipes and components upon the start of an RHR motor.

The RHR Loop A Discharge Line Low Pressure allowable value in TRM Table T3.3.2-1, item 2.b notes, is $\geq$15 psig. The licensee was unable to provide a basis for this value, but indicated the same limit had been in place for approximately 25 years since the loop select logic was removed.

Based on a review of isometric drawings and plant walkdowns, the team concluded a static head correction of approximately 39 psig should have been used to determine the correct setpoint for Switch CM-PS-270. Calculation NEDC 92-050AT, Section D.1.6.9, incorrectly calculated the head correction as 28 psig. The as-left value for Switch CM-PS-270 noted in Surveillance Procedure 6.1CSCS.305, "CSCS Discharge Piping Full Low Pressure Alarm Calibration and Functional Test (Div 1)," completed on April 26, 2001, was 53.5 psig. Subtracting the 39 psig static head correction from the 53.5 psig actual setpoint results in a value below the TRM limit. As a result of this issue, the licensee declared Switch CM-PS-270 inoperable and entered the appropriate TRM action statements.

Criterion III of Appendix B to 10 CFR Part 50 states, in part, that "[m]easures shall be established to assure that applicable regulatory requirements and the design basis . . . are correctly translated into specifications, . . . procedures, and instructions."

The team concluded the failure to correctly translate design requirements, specifically the static head correction for Switch CM-PS-270, into approved calibration procedures, was a violation of Criterion III of Appendix B to 10 CFR Part 50, and evaluated this finding in accordance with NRC Appendix B of Inspection Manual Chapter 0610*.

Using the Group 1 Questions, the team determined there was a credible impact on safety because the plant could have been operated with a void in the RHR Loop A pipe and components without control room personnel awareness, resulting in potential damage from a water hammer on pump start. Therefore, the issue was greater than a minor violation.

Using the Group 2 Questions, the team concluded the issue credibly affected the operability, availability, reliability, or function of a system or train in a mitigating system as a result of a water hammer in RHR Loop A, which is required to mitigate several design basis accidents. Therefore the significance determination process as described in NRC Inspection Manual Chapter 0609 was entered.

The team determined there was no actual loss of safety function because, when the TRM action statement was entered after the team identified this issue, RHR Loop A was vented with no significant air flow observed.

The finding did have a credible impact on safety; however, since only the mitigating systems cornerstone was affected, and the as-found value was only marginally less than the TRM allowable value, and there was no evidence that voiding had occurred, the finding is considered to be of very low safety significance (Green). The team determined the failure to properly translate design basis information into safety related calibration procedures was a violation of Criterion III of Appendix B 10 CFR Part 50. However, because of the very low safety significance, and because the licensee has included the item in their corrective action program (Notification 10089082), this design control violation is a noncited violation (50-298/0105-01) in accordance with Section VI.A of the Enforcement Policy.

.6 Safety System Testing

a. Inspection Scope

The team reviewed the program and procedures for inservice testing and inspection of the safety-related valves and pumps in the RHR system. The review included flow balancing and startup testing results; pump manufacturer pump curves; and pump and valve inservice test records.

b. Findings

No findings of significance were identified.

4 OTHER ACTIVITIES (OA)

4OA6 Management Meetings

Exit Meeting Summary

On June 22, 2001, the team leader presented the inspection results to Mr. J. Ranalli, Senior Manager of Engineering, and other members of licensee management at the conclusion of the onsite inspection. The licensee's management acknowledged the findings presented.

The team asked the licensee's management whether any materials examined during the inspection should be considered proprietary. While some material was so identified, no proprietary information is included in the report.

ATTACHMENT

KEY POINTS OF CONTACT

Licensee

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R. Church, RHR System Engineer
F. Diya, Plant Engineering Manager
S. Freborg, Staff Engineer, Engineering Support Department
W. Macecevic, Operations Manager
C. Markert, Engineering Services Department Manager
J. Ranalli, Senior Manager of Engineering
D. Reed, Operations
D. Van Der Kamp, Licensing
R. Wachowiak, Supervisor of Risk Management
A. Ward, Manager of Engineering Assessments
A. Weise, Senior Staff Engineer, Design Engineering Department
N. Wetherell, Assistant Senior Engineering Manager

NRC

A. Garcia, Engineering Associated
M. Hay, Resident Inspector
O. Tabatabai, Project Manager, License Renewal Branch

ITEMS OPENED AND CLOSED

Opened and Closed

50-298/0105-01	NCV	Inadequate measures to assure that the values used to establish the second level undervoltage relay setpoint were accurate and conservative with respect to the technical specifications. In addition, the measures for promptly identifying and correcting the adverse condition were not adequate as demonstrated by the length of time this condition has existed (since 1987). The failure to accurately translate design requirements was a violation of Criterion III of Appendix B to 10 CFR Part 50, and the untimely corrective actions was a violation of Criterion XVI of Appendix B to 10 CFR Part 50 (Section 1R21.5.b.1.).
50-298/0105-02	NCV	Failure to properly account for the static head in Calculation NEDC 92-050AT, “CM-PS-270 Setpoint Calculation,” Revision 0 (Section 1R21.5.b.2.).

DOCUMENTS REVIEWED

The following documents were selected and reviewed by the inspectors to accomplish the objectives and scope of the inspection and to support any findings:

CALCULATIONS

NUMBER	DESCRIPTION	REVISION
88-086B	Second Level Undervoltage Relay Setpoint Determination	7
92-050C	Reactor Low Pressure, Core Spray, and RHR Initiation Valve Permissive Setpoint Calculation for NBI-PS-52A2 and NBI-PS-52C2	2
DC 86-125	Removal of RHR Minimum Flow Line Orifices	01
NEDC 00-003	CNS Aux Power Load Flow and Voltage	0C2
NEDC 00-038	Containment Profiles for Steam Line Break	02
NEDC 00-049	Containment Spray Flow Rate for RHR Mode C2	01
NEDC 00-111	CNS Auxiliary Power System AC Loads	0C3
NEDC 88-190	Essential Pump Minimum Flow Damage Susceptibility	00
NEDC 89-1659	EOP Calculation 10 Single RHR Pump Injection	01
NEDC 89-1828	Maximum Flow Through the RHR Pumps	00
NEDC 91-080	RHR MOV Stroke Design Basis	3
NEDC 92-050	CM-PS-266 Setpoint Calculation	0
NEDC 92-050AT	CM-PS-270 Setpoint Calculation	0
NEDC 92-050BA	RHR-DPIS-125A & B Low RHR Pump Discharge Flow Setpoint Calculation	1
NEDC 93-008	RHR Heat Exchanger Fouling Factor Determination for Mode C2	01
NEDC 93-050	RHR Quad Temperature with Hatches Removed	02
NEDC 93-184	Verification of Senior Engineering's Calculation on the Thermal Performance of the RHR Heat Exchangers	00

CALCULATIONS

NUMBER	DESCRIPTION	REVISION
NEDC 94-034	CNS Containment Analysis	02
NEDC 94-067-017	Relief Valve RHR-RV-17RV Sizing	00
NEDC 94-067-018	Relief Valves RHR-RV-14RV & RHR-RV-15RV Sizing	00
NEDC 94-141	RHR Flow Rate for Reactor Pressure of 150 psig	00
NEDC 94-176	Radiological Dose Consequences of ECCS Leakage During a LOCA	01
NEDC 94-230	Vessel Head-Over-Drywell Capacity Curve for Input into ECCS Analysis	03
NEDC 94-231	RHR Pumps NPSH/Maximum Flow Calculation	04
NEDC 94-258	Tech. Spec. acceptance criteria for LPCI pumps flowing at 7,800 gpm	01
NEDC 95-003	Residual Heat Removal System Motor Operated Valve Data	11
NEDC 97-044	NPSH Margins for the RHR and CS pumps	01
NEDC 98-017	PC-PS-12A, B, C, D and PC-PS-101A, B, C, D Setpoints	0
NEDC 99-046	Review of GE Calculation - Cooper Nuclear Station SAFER/GESTR- LOCA Analysis, NEDC-32687P and GE-NE-L1200867-09-01	01

PLANT PROCEDURES

DOCUMENT	TITLE	REVISION
Administrative Procedure 0.5	Corrective Action Effectiveness Reviews	2
Administrative Procedure 0.27	Maintenance Rule Program	11
Alarm Response Procedure 2.3_9-3-1	Panel 9-3 - Annunciator 9-3-1 Response Procedures	2

PLANT PROCEDURES

DOCUMENT	TITLE	REVISION
Alarm Response Procedure 2.3_9-3-2	Panel 9-3 - Annunciator 9-3-2 Response Procedures	2
Alarm Response Procedure 2.3_9-3-3	Panel 9-3 - Annunciator 9-3-3	1
Emergency Operating Procedure 5.2.1	Shutdown From Outside the Control Room	26
Emergency Operating Procedure 5.3EMPWR	Emergency Power	0
Emergency Operating Procedure 5.4.3.2	Post-fire Shutdown to Mode 4 Outside Control Room	24
Emergency Operating Procedure 5.8 Attachment 1	Emergency Flowchart 1A	12
	Emergency Flowchart 2A	11
	Emergency Flowchart 2B	11
	Emergency Flowchart 3A	11
	Emergency Flowchart 5A	11
	Emergency Flowchart 6A	11 C1
	Emergency Flowchart 6B	12
	Emergency Flowchart 7A	11
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Emergency Operating Procedure 5.8 Attachment 2	EOP and SAG Graphs	11
Emergency Operating Procedure 5.8.3, Attachment 1	Alternate Rod Insertion Methods	6 C2
Emergency Operating Procedure 5.8.4	Alternate Injection Systems (Table 4)	5
Emergency Operating Procedure 5.8.6	RPV Flooding Systems (Table 6)	8
Emergency Operating Procedure 5.8.7	Primary Containment Flooding/Spray Systems	11

PLANT PROCEDURES

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791E261 Sheet 6	RHR System Elementary Diagram	N17
791E261 Sheet 7	RHR System Elementary Diagram	N06
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PROBLEM IDENTIFICATION REPORTS

1-06920	3-20229	4-00011	4-07725	4-08806	4-09273
1-18053	3-20331	4-00683	4-07864	4-08811	4-09326
2-00244	3-20476	4-01387	4-07899	4-08812	4-09327
2-13893	3-40368	4-01466	4-08041	4-09100	4-09468
2-15775	3-50334	4-01871	4-08324	4-09101	4-09543
2-24680	3-50569	4-02559	4-08332	4-09115	4-10335
2-27529	3-50570	4-02635	4-08486	4-09145	4-11234
2-28825	3-50957	4-03058	4-08791	4-09149	4-11281
3-10953	3-51446	4-04148	4-08795	4-09245	4-11539
3-20224	3-51448	4-07453	4-08805	4-09250	4-11729

4-11763	4-12044	4-12048	4-12270	4-12485	4-13777
4-11957	4-12045	4-12049	4-12271		

NOTIFICATION

10073724

NOTIFICATIONS ISSUED FOR LICENSEE SELF-ASSESSMENT

10085377	10085838	10086554	10087233	10087575
10085379	10086194	10086831	10087263	10087577
10085380	10086202	10087203	10087301	10087582
10085381	10086215	10087224	10087303	10088160
10085383	10086218	10087231	10087574	
10085415	10086268			

NOTIFICATIONS ISSUED AS RESULT OF INSPECTOR INQUIRIES

10088671	10088960	10089619	10091971	10092429
10088898	10089044	10090779	10092017	10092454
10088903	10089082	10091421	10092414	
10088959	10089401	10091890		

WORK ORDERS/WORK REQUESTS

98-0129
98-3170

GENERAL ELECTRIC NUCLEAR ENERGY DIVISION SPECIFICATIONS

NUMBER	TITLE	REVISION
21A1279	Residual Heat Removal Heat Exchanger	7
21A5790AM	RHR Pump Data Sheet	7
21A9221	Core Spray Pump and RHR Pump	5
22A1472	Residual Heat Removal System (With Steam Condensing)	0
22A1472AB	Residual Heat Removal System Data Sheet	1

SAFETY EVALUATIONS

USQE 1998-0110	CED 1998-0268, “Addition of Throttling Control Capability for RHR-MOV-M012A/B”

USQE 1999-0077	CED 1999-0012, “Automatic Closure Modifications for CS-MOV-MO5A and MO5B”
USQE 1999-0041	CED 1999-0083, “HPCI-MOV-MO58 Insulation”
USQE 2001-0011	Change to USAR Chapter X, Section 10.2.5.4, “Below Grade Areas”
USQE 2001-0016	Change to USAR Chapter VII, Section 3.0, ”Primary Containment Isolation System Control and Instrumentation”
USQE 2001-0017	Change to Technical Basis Section 3.6.1.3, “Action Bases for Remote Manual Valves”
USQE 2001-0025	Change to Procedure 5.3SBO, “Station Blackout”
USQE 2001-0019	2001-0006, “Proposed Design Change to Eliminate Pressure Locking on RHR-MOV-M039A, RHR-MOV-M039B, and RHR-MOV-M058"

SAFETY EVALUATION SCREENINGS

DOCUMENT	TITLE	REVISION
	Evaluation of Conditional Operability for Operability Evaluation OE 4-11673	
Procedure 2.2.69.1	RHR LPCI Mode	13
Procedure 5.3EMPWR/5.2.5	Emergency Power	0/37C2D
Procedure 5.8	Emergency Operating Procedures	14
Procedure 6.PRM.317	Control Room Air Sampling Electronic Calibration	8
Procedure 6.PRM.322	Containment High Range Area Monitor Channel Calibration and Setpoint Determination	8
Procedure 9.RW.9	Filling Containers with Waste/Radioactive Material	3
Procedure 10.9	Control Rod Scram Time Evaluation	39
Safety and Relief Valve Setpoint Change Request #00-018		
Setpoint Change Requests 98-43, 98-42, 98-28	Setpoint Changes for RHR-PS-105A, B, C, and D; 120A, B, C, and D; and CS-PS-37A, and B; 44A, and B	

SAFETY EVALUATION SCREENINGS

DOCUMENT	TITLE	REVISION
Software Design Change SDC 99-010		0

COMPLETED SURVEILLANCE PROCEDURES

NUMBER	TITLE	REVISION
6.2CSCS.305	CSCS Discharge Piping Full Low Pressure Alarm Calibration and Functional Test (Div 2)	May 17, 2001
6.1CSCS.305	CSCS Discharge Piping Full Low Pressure Alarm Calibration and Functional Test (Div 1)	April 26, 2001
6.1RHR.305	RHR Loop A Low Flow Switch Channel Calibration (Div 1)	May 16, 2000

MISCELLANEOUS DOCUMENTS

NUMBER	DESCRIPTION	REVISION
	Open Operator Work Around Report for Operations	June 7, 2001
	OER Document Screen for SEN 196 - Recurring Event, Inadvertent Reactor Vessel Level Decrease During Shutdown Cooling Loop Transfer	June 9, 1999
	Cooper Nuclear Station Probabilistic Risk Assessment Level 1 (IPE)	March, 1993
	Cooper Nuclear Station Probabilistic Risk Assessment Level 2 (IPE)	March, 1993
	Cooper Station-Primary Containment Leakage Rate Testing Program	0
COR002-23-02	Residual Heat Removal Lesson Student Text	17
EJ 98-141	Engineering Judgement Related to Core Spray Pump Brake Horse Power	November 18, 1998
NEDO-24708A	Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors	1

MISCELLANEOUS DOCUMENTS

NUMBER	DESCRIPTION	REVISION
NPPD letter NLS2001053	Revision of commitment date concerning submittal to resolve NRC Generic Letter 97-04 issues	May 30, 2001
Operability Evaluation 93-000-028	High flow through RHR Heat Exchangers 1A and 1B	June 21, 1993
Services Information Letter No. 175	RHR/Recirculation System Water Hammer during Primary System Cooldown	June 15, 1976
Standing Order 98-029	Operability of RHR while in Suppression Pool Cooling	September 21, 1998
Standing Order 98-006	RHR minimum flow time restrictions	November 27, 1997
Standing Order 98-004	RHR pump motor winding temperature alarm setting	September 4, 1997
STP 87-010	Measurement of Plant Electrical Loads Special Test Procedure	July 11, 1987
SWEC Letter	Engineering Evaluation of RHR System Water Hammer Occurrence of October 22, 1992	November 17, 1992
Training Manual COR002-23-02	Residual Heat Removal System Maintenance Rule Data Base	17