UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4005

January 25, 2008

Stewart B. Minahan, Vice
President-Nuclear and CNO
Nebraska Public Power District
72676 648A Avenue
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SUBJECT: COOPER NUCLEAR STATION- NRC COMPONENT DESIGN BASES INSPECTION REPORT 05000298/2007011

Dear Mr. Minahan:

On December 12, 2007, the U.S. Nuclear Regulatory Commission (NRC) completed a component design bases inspection at your Cooper Nuclear Station. The enclosed report documents our inspection findings. The inspection findings were discussed on November 2, 2007, with members of your staff. After additional in-office inspection, a final telephonic exit meeting was conducted on December 12, 2007, with a follow up discussion held on January 22, 2008, with Mr. James Flaherty and others 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. The team reviewed selected procedures and records, observed activities, and interviewed cognizant plant personnel.

This report documents six findings that were identified. The NRC has determined that violations are associated with these findings. Five of the findings were evaluated using the risk significance determination process as having very low safety significance (Green). One of the findings was subject to traditional enforcement and was characterized as Severity Level IV. Because of their very low safety significance and because they are entered into your corrective action program, these violations are being treated as noncited violations, consistent with Section VI.A of the Enforcement Policy. If you contest the subject or severity of any 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 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.

In accordance with 10 CFR 2.390 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/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

Russell L. Bywater, Chief
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Docket: 50-298
License: DPR-46

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w/Attachment: Supplemental Information

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U.S NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR-46
Report:	05000298/2007011
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	P.O. Box 98 Brownville, Nebraska
Dates:	October 1 through December 12, 2007
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Approved By:	Russell L. Bywater, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000298/2007011; October 1 through December 12, 2007; Cooper Nuclear Station; NRC Inspection Procedure 71111.21, "Component Design Basis Inspection."

The report covered a 3-week onsite period of inspection by three region-based inspectors, an Office of Nuclear Reactor Regulation Senior Reactor Analyst, and two contractors. The inspection identified six Green noncited violations. The significance of most findings is indicated by its color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process does not apply may be green or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3, dated July 2000.

A. NRC - Identified Findings

Cornerstone: Mitigating Systems

- Green: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance for the failure to verify the adequacy of design for the emergency diesel generator fuel oil system. Specifically, the licensee did not complete the necessary vortexing and net positive suction head calculations on the emergency diesel generator fuel oil storage tank and associated transfer pumps, and the fuel oil day tanks and associated booster pumps. These calculations were required to establish that adequate design margins exist to demonstrate air entrainment or cavitation does not occur during the mission time for these pumps. This finding was entered into the corrective action program under Condition Reports CNS-2007-07421 (fuel oil storage tank) and CNS-2007-07585 (fuel oil day tank).

  The finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of “Design Control.” It impacts the cornerstone objective to ensure the availability, reliability, and capability of the emergency diesel generator system to respond to initiating events and prevent undesirable consequences. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At- Power Situations," Phase 1 screening, this issue was determined to be of very low safety significance (Green) because it was determined that there was no loss of safety function. This finding has cross cutting aspects in the area of problem identification and resolution, with the Operating Experience attribute [P.2(b)]. The licensee failed to evaluate and apply various industry events associated with safety-related storage tanks vortexing into station design basis calculations. (Section 1R21.b.1)

- Green: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance for a design change, associated with the emergency diesel generator, that failed to be subjected to control measures commensurate with those applied to the original design. Specifically, a design change installed an emergency diesel generator feeder cable that could fail prior to protective device actuation on postulated asymmetrical short-circuit current values. This issue was entered into the corrective action program under Condition Report CNS-2007-07409.

The finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of “Design Control.” It impacts the cornerstone objective to ensure the availability, reliability, and capability of the emergency diesel generator system to respond to initiating events and prevent undesirable consequences. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At- Power Situations," this issue screened as having very low safety significance (Green) during a Phase 1 review because the condition did not represent a loss of system safety function. (Section 1R21.b.2)

- Green: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance, for the failure to correctly translate the emergency core cooling system design basis into instructions, procedures, and drawings. Specifically, the licensee failed to ensure design bases information was consistent within affected design documents. The licensee failed to identify that Calculation NEDC 91-078, "System Level Design Basis Review of High Pressure Coolant Injection (HPCI) System Program MOVs," and Design Calculation NEDC 98-001, "Vortex Limit for the Emergency Condensate Storage Tanks A & B," were documents that affected each other. This issue was documented in the licensee’s corrective action program as Condition Report CNS-2007-07459.

  The finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of “Design Control.” It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At- Power Situations," this issue screened as having very low safety significance (Green) during a Phase 1 review because these deficiencies were determined not to result in loss of system safety function. (Section 1R21.b.3)

- Green: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," having very low safety significance for the failure to correctly translate the design basis into specifications, drawings, procedures, and instructions. Specifically, the design criteria documents were defined as being controlled documents that provided the criteria, requirements, and bases for safety-related/important-to-safety portions of Cooper Nuclear Station. Procedure 3.32 and the related series procedures specified certain types of information to be included in the design criteria documents (i.e., logic diagrams or system templates containing system safety objectives, functional and design criteria requirements, components and parameters essential to the ability of the system to achieve its required safety functions; four different configuration matrices used to validate that current plant configuration is consistent with the design basis criteria; and various appendices, including an acceptance criteria appendix for each component, sub-system and system). The team noted during review of the design criteria documents that much of this required information was not being maintained. These issues were documented in the licensee's corrective action program as Condition Reports CNS-2007-07461 and CNS 2007-07608.

This finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of "Design Control." It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, and determined that the finding screened as very low safety significance (Green) because it was a design control deficiency confirmed not to have resulted in loss of safety function. A crosscutting aspect was identified involving the human performance component area for resources to ensure that design documentation is complete, accurate, and up-to-date (H.2(c)). (Section 1R21.b.4)

- <u>Green</u>: The team identified a noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," having very low safety significance for the failure to adequately evaluate the extent of equipment failures resulting from workmanship issues, and to determine the causes and corrective actions for this significant condition adverse to quality to prevent recurrence. During Refueling Outage 23, multiple examples of workmanship issues were identified that resulted in safety-related valve failures discovered during post-maintenance testing. Subsequent to the implementation of corrective actions to address this issue, a directly related workmanship condition was identified involving Safety-Related Valve HPCI-MOV-MO16. This valve was returned to service, for approximately 10 months, before identifying that a nonconforming condition involving workmanship existed that required correction prior to returning the valve to service. The licensee entered this condition into their corrective action program as Condition Report CNS-2007-07609.

  This finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of "Equipment Performance." It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to evaluate the extent of condition for the valves, which were potentially affected, and to determine the causes for the multiple workmanship issues. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," this issue screened as having very low safety significance during a Phase 1 review because the valve workmanship issues were corrected prior to returning to service with the exception of one valve, which was determined to be functional in the nonconforming condition. The cause of this finding had crosscutting aspects associated with problem identification and resolution, related to the Corrective Action Program attribute [P.1.(c)], for thoroughly evaluating problems. The resolutions address causes and extent of conditions, as necessary. (Section 1R21.b.5)

- SL IV: The team identified a noncited Severity Level IV violation for the failure to comply with the requirements of 10 CFR 50.71(e). The correct value for the automatic depressurization system accumulator minimum pressure was not used to revise the Updated Safety Analysis Report. Specifically, the licensee's technical specifications and Design Calculation NEDC 88-306 require a minimum of 88 psig to assure five actuations of the safety relief valves with the drywell at atmospheric conditions. The Updated Safety Analysis Report lists a minimum pressure of 68.6 psig for this function. The Updated Safety Analysis Report stated pressure of 68.6 psig was incorporated as part of the licensee's Updated Safety Analysis Report rebase line project and became effective on March 10, 2000. The licensee was unable to provide a basis for the lower pressure stated in the Updated Safety Analysis Report.

  This violation was subject to traditional enforcement because it had the potential to impact the regulatory process. This finding is considered more than minor because use of this lower pressure value could render the automatic depressurization feature incapable of performing its design function. In accordance with NRC Enforcement Policy, the NRC has concluded that this is a Severity Level IV violation. Because this violation was of very low safety significance, was not repetitive or willful, and it was entered into the licensee's corrective action program as Condition Report CNS-2007-07468, this violation is being treated as an noncited violation, consistent with Section VI.A.1 of the NRC Enforcement Policy. (Section 1R21.b.6)

REPORT DETAILS

1. REACTOR SAFETY

Inspection of component design bases verifies the initial design and subsequent modifications and provides monitoring of the capability of the selected components and operator actions to perform their design bases functions. As plants age, their design bases may be difficult to determine and an important design feature may be altered or disabled during a modification. The plant risk assessment model assumes the capability of safety systems and components to perform their intended safety function successfully. This inspectable area verifies aspects of the Initiating Events, Mitigating Systems and Barrier Integrity cornerstones for which there are no indicators to measure performance.

In addition to performing the baseline inspection, the team reviewed actions taken by the licensee in response to previously identified significant issues associated with engineering performance.

1R21 Component Design Bases Inspection (71111.21)

The team selected risk-significant components and operator actions for review using information contained in the licensee's probabilistic risk assessment. In general, this included components and operator actions that had a risk achievement worth factor greater than two or Fussel-Vesely importance value greater than 1E-4.

a. Inspection Scope

To verify that the selected components would function as required, the team reviewed design basis assumptions, calculations, and procedures. In some instances, the team performed independent calculations to verify the appropriateness of the licensee engineers' conclusions. The team also verified that the condition of the components was consistent with the design bases and that the tested capabilities met the required criteria.

The team reviewed maintenance work records, corrective action documents, and industry operating experience information to verify that licensee personnel considered degraded conditions and their impact on the components. The team determined the area and scope of the inspection by reviewing the licensee’s probabilistic risk analysis models to identify the most risk significant systems, structures, and components according to their ranking and potential contribution to dominant accident sequences and/or initiators. The team also used a deterministic effort in the selection process by considering recent inspection history, recent problem area history, and all modifications developed and implemented.

For operator actions, the team performed a detailed review of five risk-significant, time critical operator actions (five samples). These actions were selected from the licensee’s probabilistic risk assessment rankings of human action importance based on risk achievement worth values. The time-critical operator actions were determined by a review of the ratio of "time available" to "time required" to perform the specific operator actions. The time available to perform the operator actions is calculated by plant-specific thermal-hydraulic analyses using the modular accident analysis program code, and the time required to perform the specific operator actions is based on plant simulator response data and operator interviews providing estimated times for diagnosis and manipulation actions. For the selected operator actions, the team observed simulator performance of associated procedures with a selected plant operations crew to assess operators' knowledge level, adequacy of procedures, and use of any special equipment required. Additionally, the team observed in-plant simulated performance of risk significant operator recovery actions by station operators during the simulator scenarios.

For each of the selected components, the team assessed the adequacy of calculations, analyses, engineering processes, and engineering and operating practices that were used by the licensee to support the performance of the component selected for review and the necessary support systems during normal, abnormal, and accident conditions. Acceptance criteria utilized by the NRC inspection team included NRC regulations, the technical specifications, applicable sections of the Final Safety Analysis Report, applicable industry codes and standards, as well as, industry initiatives implemented by the licensee's programs.

The team also performed a margin assessment and detailed review of the selected risk significant components to verify that the design bases have been correctly implemented and maintained. This design margin assessment considered original design issues, margin reductions due to modification, or margin reductions identified as a result of material condition issues. The margin review also included all conditions that could reasonably cause loss of selected component function. Equipment reliability issues were also considered in the selection of components for detailed review. These included items, such as failed performance test results; significant corrective actions; repeated maintenance; 10 CFR 50.65(a)1 status; operable, but degraded, conditions; NRC resident inspector input of problem equipment; system health reports; industry operating experience; and licensee problem equipment lists. Consideration was also given to the uniqueness and complexity of the design, operating experience, and the available defense in depth margins.

The components selected for review were:

- Emergency Condensate Storage Tank A/B
- 4.16kV ESF Bus G
- Undervoltage/Secondary Level Undervoltage Relays 27x3/1F
- Auto Transformer 345/161kV
- 125 Vdc Battery and Charger Train B
- Diesel generator voltage regulator control circuit
- Automatic depressuration system accumulators
- Service Water Booster Pumps 1A/1C
- Service Water Motor Operated Valves MO-89/2797
- Low Pressure Coolant Injection Pump A
- Residual heat removal heat exchanger
- Residual Heat Removal Motor-Operated Valve MO-16/39
- Primary Containment Vents Air-Operated Valves 237/245
- Hydraulic Control Unit Scram Pilot Solenoid-Operated Valve SW-2797A

- High pressure coolant injection pump
- High Pressure Coolant Injection motor-operated valve, 14/15
- Service water strainer
- Reactor Core Isolation Cooling motor-operated valves
- Emergency diesel generator fuel oil pumps

The operator actions reviewed were:

- Actions to depressurize the reactor pressure vessel during a transient, stuck-open/cycling relief valve, loss of offsite power, or station blackout events;

- Actions to control high-pressure coolant injection flow to prevent reaching high reactor pressure vessel water level;

- Actions to restore Emergency Diesel Generator 2 during loss of offsite power, or station blackout events with high pressure coolant injection and reactor core isolation cooling failures;

- Actions to restore service water pump after a loss of offsite power event; and

- Actions to manually start low-pressure injection flow in response to an automatic start failure during a loss of offsite power, or station blackout event.

The operating experience issues reviewed were:

- NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment"

- NRC Generic Letter 89-10 – 10 CFR 50.54(f), "Safety-Related Motor-Operated Valve Testing and Surveillance"

- NRC Generic Letter 96-05, "Periodic Verification of Design-Basis Capability of Safety-Related Motor-Operated Valves"

- Information Notice 2006-22, "New Ultra-Low Sulfur Diesel Fuel Oil Impact on EDG Performance"

- GE Part 21, "Crush Pressure for Suction Strainer Analysis"

- Information Notice 2007-27, "Recurring Events Involving EDG Operability"

- NRC Event Notification Report for January 5, 2006, Event Number 42242, discusses the potential for vortex formation at the suction of the emergency diesel generator fuel oil transfer pumps of Callaway Plant

b. Findings

b.1 Failure To Consider Vortexing and Available Net Positive Suction Head Impact on the Emergency Diesel Fuel Oil System

Introduction: The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to establish measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures and instructions. Specifically, the licensee failed to perform an evaluation of the potential for vortex formation of the two diesel fuel oil storage tanks and the two day tanks and net positive suction head of the associated pumps. Depending on the magnitude of air entrainment, such vortex formation could be detrimental to the operation of the diesel fuel transfer pumps and booster pumps and could lead to their malfunction.

Description: Fuel oil transfer pumps and booster pumps are components of the diesel fuel oil system and were considered part of the inspection team review. Associated calculations are important to ensure that the diesel generator has adequate volume of fuel available following a design basis event and that adequate margin is available to prevent air entrainment when fuel levels near usable capacity.

The team requested the licensee provide calculations for potential vortex formation at the two fuel oil storage tanks, as well as, the net positive suction head available for the diesel fuel oil transfer pumps. The required net positive suction head and vortexing calculations were not available when the team requested them. To document this issue, Condition Report CNS- 2007-07421 was generated. The licensee, on October 26, 2007, began to prepare Calculation NEDC 07-090 on the fuel oil storage tanks. The team later requested the licensee provide calculations for potential vortex development at the two Cooper Nuclear Station fuel oil day tanks, as well as the available net positive suction head for the diesel fuel oil booster pumps and engine driven pumps. In response to this request, the licensee generated Condition Report CNS-2007-07585. Preliminary results identified that the available net positive suction head was adequate and that margin existed so that vortexing would not occur, but there was little margin for the required 49,500 gallons of fuel for a 7-day run. The team reviewed the results as part of the in-office inspection that continued following the team leaving the site. The licensee added the final calculation to NEDC 07-090, which was approved on December 4, 2007.

The licensee was presented various opportunities to complete a proper evaluation of the emergency diesel fuel oil tanks. For instance; the licensee evaluation in Condition Report CNS-2006-09585 addressed vortexing in many essential tanks as part of their evaluation of Information Notice 2006-21, "Operating Experience Regarding Entrainment of Air into the Emergency Cooling and Containment Spray Systems," but did not address the emergency diesel system fuel oil tanks. In addition, there was the Operating Experience of the Callaway Plant Event 42242, which the NRC notified licensees that their preliminary evaluation on the potential for vortex formation on the existing technical specification level requirements for the Callaway Plant emergency diesel generator underground fuel oil storage tanks may be non-conservative.

Analysis: This finding is a performance deficiency because the licensee did not properly evaluate and document that essential components needed to support the emergency diesel generator operation following a design basis accident had been appropriately evaluated for vortexing or available net positive suction head. This finding was determined to be more than minor because it was associated with the design control attribute of the mitigating systems cornerstone, and it affected the cornerstone objective to ensure the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At-Power Situations," Phase 1 screening, this issue was determined to be of very low safety significance (Green) because it was determined that there was no loss of safety function. This finding had crosscutting aspect in the area of problem identification and resolution with the Operating Experience attribute [P.2(b)]. The licensee failed to evaluate and apply various industry events associated with safety-related storage tanks vortexing into station design basis calculations.

Enforcement: Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that the applicable design bases are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents. Contrary to the above, the measures established to assure the applicable design bases are correctly transferred in plant documents were not adequate. Specifically, licensee engineers failed to translate design requirements (i.e., vortex or available net positive suction head prevention) into design basis calculation. Because this violation is of very low safety significance and was entered into the licensee's corrective action program as Condition Reports CNS-2007-07421 and CNS-2007-07585, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2007011-01, Failure To Consider Vortexing and Available Net Positive Suction Head Impact on the Emergency Diesel Fuel Oil System.

b.2. Installation Of Essential Electrical Cable With Inadequate Fault Current Ratings And Not In Accordance With Original Design Basis Requirements

Introduction: The team identified a Green noncited violation for the failure to comply with 10 CFR Part 50, Appendix B, Criterion III, "Design Control." Specifically, a design change installed electrical cabling (DG125) with inadequate fault current ratings in the Emergency Diesel Generator 2 system. The licensee correctly identified the cable as inadequate in a subsequent design calculation, but failed to recognize that the smaller cable did not meet the sizing requirements (2/0 American Wire Gauge (AWG)) of the original design. Also, the corresponding essential cable installed in Emergency Diesel Generator 1 system (DG180) is not analyzed in the design calculation that determines the cable short-circuit withstand ratings.

Description: Design Calculation NEDC 91-190, "AC Equipment and Cable Short Circuit Withstand Ratings," lists eight electrical cables that are not adequately sized to withstand the potential short circuit current to which they may be subjected during fault conditions. One cable, DG125, "Feeder from line side of EG2 Breaker to MCC-DG2 Transformer 1E," is classified as an essential cable and is listed as Size 6 AWG. Failure of Cable DG125 would render diesel support systems inoperable, which would render the emergency diesel generator inoperable.

The conductors in Cable DG125 were manufactured by the Okonite Company. The continuous current rating of a Size 6 AWG conductor, based on ampacity data from Okonite tables, is approximately 75 amperes at a 90° C cable temperature. The steady-state loading on Cable DG125 is less than this limit, indicating that the cable is adequately sized to meet its steady-state loading requirement. However, the calculated symmetrical short-circuit fault current is 31,290 amps and the calculated short-circuit withstand time for Cable DG125, at the symmetrical current value is 0.0036 seconds (3.6 ms). Further, the analysis shows that Cable DG125 could be exposed to an asymmetrical fault current of 57,280 amperes for ½-cycle (8.3 ms). A withstand time is not calculated for this asymmetrical current value.

The protective device for this cable is listed in Design Calculation NEDC 91-190 as 4kV Breaker 1GE, with an overcurrent rating setpoint of 47,424 amperes at 0.01 seconds. A review of Design Calculation NEDC 86-105B, "CNS Critical AC Bus Coordination Study," Revision 7C12, page 19, sheet 6, confirms that Breaker 1GE will trip at 47,000 amperes in approximately 0.16 seconds. The time-current characteristic curve on page 19, sheet 6, is not analyzed at times <0.1 seconds. Design Calculation NEDC 86-105B indicates that the conductors are also protected by Bussmann-type JCY50E (E-rated, 50 Ampere) fuses.

The time-current characteristic curve for this type of fuse provides no information for fault exposure times <0.01 seconds. The licensee provided additional information, which indicates this fuse would limit the current if the symmetrical faulted condition lasted more than ½-cycle (8.33 ms), but does not consider the asymmetrical fault current condition. Since the fuse characteristics cannot be verified for interrupting times close to 3.6 ms, a potential exists for the cable to fail before the protective devices will actuate.

The design basis requirement for cable sizes in the 4160V system, based on Burns and Roe Calculation 2.05.06, requires a minimum cable size of 2/0 for solidly grounded systems. This sizing is based on a short-circuit value, corrected for cable temperature of 90° C and dc current offset of 29,000 amps. The current rating of a 2/0 AWG cable, based on ampacity data from Okonite tables, is approximately 204 amperes at a 90° C cable temperature. A 2/0 AWG cable will withstand approximately 40,000 amperes for 3 cycles. This sizing difference between the existing Size 6 AWG and a 2/0 AWG cable results in a reduction in steady-state current margin of approximately 63 percent. The licensee identified that a failure of Transformer 1E would potentially cause this cable to fail. Failure of both cable and transformer could extend the time the emergency diesel generator is inoperable, as opposed to only the transformer failing with a larger capacity cable.

Cable DG180 is shown in Calculation NEDC 86-105B, "CNS Critical AC Bus Coordination Study," Revision 7C12, as a 2/0 AWG cable. This essential cable performs the same function on the Emergency Diesel Generator 1 system as Cable DG125 performs on the Emergency Diesel Generator 2 system. Both cables have similar loadings and current requirements; however, Cable DG180 is not analyzed in Cooper Nuclear Station Design Calculation NEDC 91-190. The licensee stated that

Cable DG180 was not mentioned or analyzed in Design Calculation NEDC 91-190 because of engineering judgment used in the modification package that installed the cable.

Analysis: The failure to subject design changes to the same control measures as the original design and ensure that essential components are included in the appropriate analysis is a performance deficiency. The team determined the finding is more than minor because it affects the design control attribute of the mitigating system cornerstone objective to ensure the availability, reliability, and capability of the emergency diesel generator system to respond to initiating events and prevent undesirable consequences. In this case Design Calculation NEDC 91-190 results showed Cable DG125 was undersized, the licensee did not recognize that the design failed to meet the original design basis requirement for cable sizing, and the licensee did not recognize that Cable DG180 was not analyzed in the design calculation.

Using Inspection Manual Chapter 0609, "Significance Determination Process," Phase 1 worksheet, the team determined that this finding is of very low safety significance (Green) because the condition did not represent a loss of system safety function.

Enforcement: Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, "Design Control," requires, in part, that design changes be subject to design control measures commensurate with those applied to the original design. Criterion III also requires that design control measures shall provide for verifying or checking the adequacy of the design, such as by the performance of design reviews, by the use of alternate or simplified calculational methods, or by the performance of a suitable testing program.

Contrary to these requirements, the installation of Cable DG125 as a Size 6 AWG did not meet minimum cable size requirements for connection to the 4160Vac system, as required in the original design basis, and Cable DG180 was not included in a design calculation. Because this finding is of very low safety significance and has been entered in the licensee's corrective action program as Condition Report CNS-2007-07409, this violation is being treated as a noncited violation consistent with Section VI.A of the NRC Enforcement Policy: NCV 05000298/2007011-02, Installation of Essential Electrical Cable with Inadequate Fault Current Ratings and Not in Accordance with Original Design Basis Requirements.

b.3. Failure To Ensure that Design Bases Information Remains Consistent Within Affected Design Documents

Introduction: The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure that important design bases information would remain consistent within affected design documents. Specifically, the licensee failed to identify that Calculation NEDC 91-078 "System Level Design Basis Review of High Pressure Coolant Injection (HPCI) System Program MOVs," and Design Calculation NEDC 98-001, "Vortex Limit for the Emergency Condensate Storage Tanks A & B," were documents that affected each other.

Description: During review of the licensee's design calculations and related design documents to verify that motor-operated valve in-service test acceptance criteria were consistent with design requirements, the team noted that the design basis stroke times established in Revision 1 to Design Calculation NEDC 98-001, dated May 1, 2001, for Motor-Operated Valves HPCI-MOV-MO17 and -M058, had not been incorporated into Revision 3 to Calculation NEDC 91-078 dated September 10, 2002.

Calculation NEDC 98-001 evaluated the emergency condensate storage tanks as viable water supply sources for high pressure coolant injection and reactor core isolation cooling systems, and to ensure that no vortexing/air entrainment conditions will exist. Assumptions are made for stroke times of Motor-Operated Valves HPCI-MOV-MO17 and -MO58 (the high pressure coolant injection pump suction from the emergency condensate storage tank and suppression pool, respectively) and these assumptions are correlated to necessary emergency condensate storage tank water levels to avoid vortexing. The assumptions established a design basis stroke time (Motor-Operated Valves HPCI-MOV-MO17 ≤ to 78 seconds and HPCI-MOV-MO58 ≤ to 82 seconds) that must be controlled and incorporated in all other affected lower-tier design documents. During review of Calculation NEDC 91-078, the team noted that Section 4.4 stated that there was a passive open safety function and an active close safety function for Motor-Operated Valve HPCI-MOV-MO17. Further, Sections 4.4.2.5 and 4.4.3.5 stated, respectively, that there was no specified design basis opening or closing stroke times for Motor-Operated Valve HPCI-MOV-MO17. Similarly, for Motor-Operated Valve HPCI-MOV-MO58, Section 4.10 stated that there were active safety functions to both open and close. Sections 4.10.2.5 and 4.10.3.5, respectively, stated that there was no specified design basis opening or closing stroke times for Motor-Operated Valve HPCI-MOV-MO58. The issue was documented in the licensee's corrective action program as Condition Report CNS-2007-07459.

Analysis: The licensee's failure to ensure that important design bases information would remain consistent within affected design documents was a performance deficiency. The team determined that the performance deficiency was more than minor because it is associated with the Mitigating Systems Cornerstone attribute of "Design Control." It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, inclusion of the inaccurate design basis information into the affected design documents could have resulted in a failure to establish appropriate in-service test acceptance criteria, thus, allowing a component to not meet its design requirements. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At- Power Situations," Phase 1 screening, this issue screened as having very low safety significance because these deficiencies were determined not to result in loss of system safety function.

Enforcement: Criterion III of Appendix B to 10 CFR Part 50 states, in part, that measures shall be established to assure that applicable regulatory requirements and design bases for those structures, systems, and components are correctly translated into procedures and instructions. The design control measures shall provide for verifying or checking the adequacy of design. Measures shall be established for the identification and control of design interfaces for coordination among participating design organizations. These measures shall include the establishment of procedures for the reviews, approval, release, distribution, and revision of documents involving design interfaces.

Section 3 in Revision 7 to Engineering Procedure 3.1 defines an "Affected Document" as a design output document that requires revision as the result of the design process, and "Design Process" as the documented design practices such as calculations, analyses, evaluations, or other documented engineering activities that substantiate the final design.

Section 4 in Revision 28 to Engineering Procedure 3.4.7, "Design Calculations," requires that all affected documents are correctly identified in the design calculation's cross-reference index form and that the listing is complete.

Contrary to the above, the licensee's design control measures failed to correctly identify an affected document in the cross-reference index of a design calculation, thus, important design information regarding opening and closing stroke times of Motor-Operated Valves HPCI-MOV-MO17 and -MO58 (the high pressure coolant injection pump suction from the emergency condensate storage tank and suppression pool, respectively) was not being maintained consistent within applicable design documents. The assumptions that established the stroke times are correlated to necessary emergency condensate storage tank water levels to avoid vortexing. Because this finding is of very low safety significance and was entered into the licensee's corrective action program as Condition Report CNS-2007-07459, this violation is being treated as a noncited violation, consistent Section VI.A.1 of the NRC Enforcement Policy: NCV 05000298/2007011-03, Failure To Ensure that Design Bases Information Remains Consistent Within Affected Design Documents.

b.4. <u>Failure To Comply With Design Control Program Requirements</u>

<u>Introduction</u>: The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion III, "Design Control," for failure to ensure that required design bases information was being appropriately maintained in topical design and system-level design criteria documents. Specifically, the licensee failed to comply with design control program requirements by not including procedurally identified and required design information appendices, matrices, and logic diagrams in the design criteria documents.

<u>Description</u>: During review of the licensee's design basis program (described in Procedure 3.32, "Design Basis Program Description"), the team noted the existence of design criteria documents. Procedure 3.32 defined design criteria documents as being controlled documents that provide the criteria, requirements, and bases for the design of a portion of Cooper Nuclear Station. The procedure also stated that a design criteria document has specific uses, including: (1) provides an aid to personnel who develop configuration change packages, write safety evaluations, and perform system operational evaluations and design/set point reviews; (2) provides a primary reference source for anyone needing a basic understanding of the design portion of the plant; and (3) provides an input to operations personnel during development of operating instructions and training of personnel. Procedure 3.32 also stated that design criteria documents are to be updated, maintained, and controlled in accordance with the requirements of the Engineering Procedures 3.32 series.

The team requested the design criteria documents applicable to the components/ systems selected for evaluation. While Procedure 3.32 specifically identified 26 design criteria documents, the licensee provided approximately 40 design criteria documents for the team use. The team reviewed the Engineering Procedures 3.32 series to identify the specific information required to be maintained in the design criteria documents.

Procedure EP 3.32.4, "Logic Diagram Development," states the purpose is to provide guidance for developing a logic diagram. The logic diagram is intended to be a template for the system or topic which represents those components, parameters, requirements, and criteria that are critical to the safety functions of the system or topic.

Section 2.3 states that the system template contain the system safety objective, functional and design criteria requirements, and components and parameters, which are essential to the ability of the system to achieve its required safety functions.

The introduction in the design criteria document states that the design criteria document was developed using a template approach. The template is the system logic diagram shown in Appendix A, and identifies the system functional criteria that are essential to nuclear safety, and the additional system design criteria imposed to ensure that the system could perform its required functions under worst-case expected transient and postulated conditions. The logic diagram of Appendix A is important because it identifies the functional and design aspects of the system, component and structures which then must be addressed in the design criteria document.

Procedure EP 3.32.5, "Configuration Matrix Development," states that the purpose is to provide the design engineering group with a set of specific and general guidelines for developing configuration matrices.

Section 2 states that configuration matrices are used in validating that the current plant configuration is consistent with the design basis criteria and requirements documented in the design criteria documents. Further, it states that there are typically four types of configuration matrices used in a system design criteria documents, including: component matrix; setpoint matrix; licensing commitment matrix; and description/requirement matrix.

Section 4 states that each component described in Appendix B of the design criteria documents shall be listed in the component configuration matrix.

Section 5 states that the instruments and other components (such as relays) in the system with important set points shall be listed in the set point configuration matrix.

Section 6 states that each design criteria document shall have a licensing commitment matrix.

Section 7 states that each design criteria documents shall have a description/ requirement matrix prepared.

Procedure EP 3.32.8, "DCD Verification and Validation," in Section 7, states that by using Attachment 3, the assigned verifier(s) shall document verification of Appendices C through I of the design criteria documents; thereby ensuring that they have been completely, accurately, and correctly translated from the source documents.

Section 14 states that the Acceptance Criteria, Appendix L, to the design criteria documents for each component, sub-system, and system within a system shall specify and verify testing, operability, procurement, testing frequencies, and other surveillance recommendations. It further states that the Acceptance Criteria, Appendix L, is a part of the design criteria documents and shall be incorporated into the design criteria documents as Appendix L.

During the team review of the design criteria documents applicable to the components/systems selected for evaluation, the following issues were identified:

(a) None of the reviewed design criteria documents contained Appendix A. In addition to being mentioned in the Introduction, Appendix A is also referenced in several follow-on chapters in each design criteria document. (The licensee initiated Condition Report CNS-2007-07461 to address this specific issue.)

(b) Of the design criteria documents reviewed by the team, only one of the four required matrices was found, which was the licensing commitment matrix.

(c) None of the reviewed design criteria documents contained Appendix G.

(d) None of the reviewed design criteria documents contained Appendix L, Acceptance Criteria.

Issues (b), (c), and (d) were documented in the licensee's corrective action program as Condition Report CNS-2007-07608.

Analysis: The team determined that the performance deficiency was more than minor because it is associated with the Mitigating Systems Cornerstone attribute of "Design Control". It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. The team evaluated the finding using Inspection Manual Chapter 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," and determined that the finding screened as very low safety significance (Green) because it was a design control deficiency confirmed not to have resulted in loss of safety function. A crosscutting aspect was identified for this finding involving the human performance component area for resources to ensure that design documentation is complete, accurate, and up-to-date (H.2(c)).

Enforcement: Part 50 of Title 10 of the Code of Federal Regulations, Appendix B, Criterion III, "Design Control," requires, in part, that measures shall be established to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. These measures shall include provisions to assure that appropriate quality standards are specified and included in design documents and that deviations from such standards are controlled.

Measures shall be established for the identification and control of design interfaces for coordination among participating design organizations. These measures shall include the establishment of procedures for the reviews, approval, release, distribution, and revision of documents involving design interfaces.

Contrary to the above, the licensee failed to update, maintain, or control design criteria documents. Specifically, design criteria documents did not include the required system logic diagrams, three of the required configuration matrices, and two of the required appendices.

Because the violation was of very low safety significance and the licensee entered the finding into their corrective action program as Condition Reports CNS-2007-07461 and CNS-2007-07608, this violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC Enforcement Policy: NCV 05000298/2007011-04, Failure to Comply with Design Control Program Requirements.

b.5. Inadequate Corrective Actions Associated With Multiple Workmanship Issues On Safety-Related Valves

Introduction: The team identified a Green noncited violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Actions," for the failure to adequately evaluate the extent of equipment failures resulting from workmanship issues and to determine the causes for this significant condition adverse to quality to prevent recurrence. During Refueling Outage 23, multiple examples of workmanship issues were identified that resulted in safety-related valve failures discovered during post-maintenance testing. Subsequent to the implementation of corrective actions to address this issue, a directly related workmanship condition was identified involving safety-related Motor-Operated Valve HPCI-MOV-MO16. This valve was returned to service for approximately 10-months before identifying that a nonconforming condition involving workmanship existed that required correction prior to returning the valve to service. The licensee entered this condition into their correction action program as Condition Report CNS-2007-7609.

Description: The team reviewed recent maintenance activities and operational events associated with selected high-risk, low-margin, motor-operated valves. As part of the component selection process, the team selected the high pressure coolant injection system Motor-Operated Valve MO16, steam supply outboard isolation valve on the High Pressure Coolant Injection Pump 2. One of these activities included Maintenance Work Order 4406219, dated October 24, 2006, in which Motor-Operated Valve HPCI-MOV-MO16 was refurbished and post-maintenance testing was completed on November 20, 2006. This activity was performed in conjunction with the repair of multiple motor-operated valves during Refueling Outage 23, completed in the latter part of November 2006.

On September 17, 2007, Motor-Operated Valve HPCI-MOV-MO16 was removed from service to perform preventive maintenance in accordance with Maintenance Work Order 4498763. During the performance of this activity, it was discovered that although the procedural step to fill the housing, ". . . until the worm gear is totally immersed in grease," had been checked as complete during the refurbishment of Motor-Operated Valve HPCI-MOV-MO16 during Refueling Outage 23, the clutch housing had not been filled with grease as required by Maintenance Procedure 7.5.12, "SMB-O Through SMB-4 MOV Refurbishment." This deficiency was documented in Condition Report CNS-2007-06386, dated September 17, 2007, as a "workmanship" issue. The operability determination associated with this condition, performed in accordance with Procedure ENN-OP-104, "Operability Determinations," concluded that, as a result of the normally open configuration of this valve which is cycled quarterly and the absence of observable gear degradation, Motor-Operated Valve HPCI-MO-MO16 remained operable. The extent of condition associated with Condition Report CNS-2007-06386, also documented that four other motor operated valves were impacted by the workmanship concerns, " … that fit the criteria for checking the grease level in the clutch housing." However, only one of these valves, RHR-MO-MO25A (RHR Loop A Injection Inboard Isolation) was verified to have grease in the clutch housing. The remaining actions to confirm the level of grease in valves RCIC-MO-MO21 (RCIC Injection to Reactor) and RR-MO-MO53 (Reactor Recirculation Pump B, Discharge) were deferred until the next refueling outage (RFO 24) and Valve CD-MO-90MV (Feedwater Heater A-4 Vent) had no work action assigned. As noted in Condition Report CNS-2007-06386, valves RCIC-MO-MO21 and RHR MO-MO25A are cycled quarterly during valve operability testing and Valve CD-MO-90MV was classified as a non-essential component.

The team also reviewed the conditions documented in Condition Report CNS-2006-09839, dated November 28, 2006, which identified multiple deficiencies involving motor-operated valves refurbished during Refueling Outage 23. In particular, the following post-maintenance motor-operated valve deficiencies, which involved the same contractor maintenance personnel that worked on Motor-Operated Valve HPCI-MOV-MO16, were documented:

- CD-MOV-MO68; MO limit switch remained de-clutched,
- CD-MOV-MO90; close limit switches installed 180 degrees off,
- RF-MOV-MO29; torque switch had fallen apart preventing valve closure,
- RHR-MOV-MO25A; bonnet cocked/valve operation would have damaged valve stem,
- CW-MOV-MO102; operator misalignment resulted in valve disc going through seat,
- HPCI-MOV-MO15; incorrect torque switch setting resulted in over-thrust condition,
- HV-MOV-MO272; following rework, excessive gap identified between seat ring and disc,
- AR-MOV-MO163; improperly set limit switch resulted in motor failure, and
- MS-MOV-MO77; improperly installed torque switch resulted in over-thrust condition.

The apparent cause for the deficiencies identified in Condition Report 2006-09839, classified as a Category B condition, concluded that these motor-operated valve failures were attributable to "workmanship" issues and inadequate procedural controls. However, the corrective actions associated with Condition Report CNS-2006-09839, were limited to the specific valves that failed post-maintenance testing. As a result of this limitation, the licensee failed to establish the means by which they could have; (1) identified the missing grease in the clutch housing of Motor-Operated Valve HPCI-MOV-MO16, which was refurbished during Refueling Outage 23 by the same contractors, and (2) completed the verification of the grease level in the four other valves identified in the extent of condition for Condition Report CNS-2007-06386. The team was not presented with any corrective action documents that addressed the causes for the workmanship issues or any extent of condition review that justified acceptance of other valves the contractors had worked that had been returned to service.

The team also found the licensee's operability determination associated with Condition Report CNS-2007-06386, lacked detail and failed to consider the as-found nonconforming condition with Motor-Operated Valve HPCI-MOV-MO16 and the potential impact on its ability to close under accident conditions without the valve gears having been lubricated in accordance with the vendor instructions. Specifically, the lack of lubrication would result in changes to the performance characteristics of the valve.

Analysis: This finding is more than minor because it is associated with the Mitigating Systems cornerstone attribute of "Equipment Performance." It impacts the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the licensee failed to evaluate which valves were potentially affected and to determine the causes for the multiple workmanship issues. Using Manual Chapter 0609, Appendix A, "Determining the Significance of Reactor Inspection Findings for At- Power Situations," this issue screened as having very low safety significance during a Phase 1 review because the valve workmanship issues were corrected prior to returning service with the exception of one valve, which was determined to be functional in the nonconforming condition. The cause of this finding had crosscutting aspects associated with the problem identification and resolution, related to the Corrective Action Program attribute [P.1.(c)], for thoroughly evaluating problems such that the resolutions address causes and extent of conditions, as necessary.

Enforcement: Criterion XVI of Appendix B, to 10 CFR Part 50, states, "Measures shall be established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances are promptly identified and corrected. In the case of significant conditions adverse to quality, the measures shall assure that the cause of the condition is determined and corrective action taken to preclude repetition.

Contrary to the above, the licensee failed to adequately evaluate the extent of equipment failures resulting from workmanship issues and to determine the causes for this significant condition adverse to quality to prevent recurrence. During Refueling Outage 23, multiple examples of workmanship issues were identified that resulted in safety-related valve failures discovered during post-maintenance testing. Subsequent to the implementation of corrective actions to address this issue, a directly related workmanship condition was identified involving safety-related Motor-Operated Valve HPCI-MOV-MO16. This valve was returned to service for approximately 10 months, before identifying that a nonconforming condition involving workmanship existed that required correction prior to returning the valve to service. Because the violation was of very low safety significance and the licensee entered the finding into their corrective action program, the violation is being treated as a noncited violation consistent with Section VI.A.1 of the NRC enforcement policy: NCV 05000298/2007011-05, Inadequate Corrective Actions Associated With Multiple Workmanship Issues On Safety-Related Valves.

b.6 Failure To Comply With The Requirements Of 10CFR50.71(e) and To Assure The Updated Safety Analysis Report Has The Latest Information Developed

Introduction: The team identified a noncited Severity Level IV violation of 10 CFR 50.71(e) requirements to periodically update the Final Safety Analysis Report to assure that the information included in the report contains the latest information developed.

Description: The automatic depressurization feature of the pressure relief system serves to back up the high pressure coolant injection system under loss-of-coolant accident conditions. If the high pressure coolant injection system fails to operate, and one of the low pressure coolant injection or core spray pumps is available, the nuclear system is depressurized by the automatic depressurization system relief valves to permit the low pressure coolant injection and core spray systems to inject water, protecting the fuel barrier.

Cooper Nuclear Station's automatic depressurization system feature includes six relief valves, each equipped with an accumulator. The six automatic depressurization system accumulators are tested to ensure that they will provide sufficient motive force to actuate the relief valves at least five times at atmospheric drywell pressure after being isolated from the nitrogen supply for 1 hour. Meeting this test condition satisfies the design requirement of two valve actuations with the drywell at 70 percent of design pressure. Cooper Nuclear Station Technical Specifications and Design Calculation NEDC 88-306, "Analysis of ADS Accumulators," Revision 0, state that the minimum accumulator pressure required assuring five actuations of the relief valves, with the drywell at atmospheric conditions, is 88 psig. This calculation was approved on January 10, 1989, and subsequently incorporated into the design basis. The Updated Safety Analysis Report IV-4, Section 4.6, "Safety Evaluation," incorrectly indicates that a minimum pressure of 68.6 psig is required to assure five relief valve actuations under the test conditions, which was incorporated as part of licensee's Updated Safety Analysis Report rebaseline project and became effective on March 10, 2000.

Analysis. This violation was subject to traditional enforcement because it had the potential to impact the regulatory process. This finding is considered more than minor because use of this lower pressure value could render the automatic depressurization feature incapable of performing its design function. The NRC characterized the violation as Severity Level IV because the failure to update the Updated Safety Analysis Report did not impede or influence regulatory action related to changes made to the facility, or the NRC's review of proposed license amendments.

Enforcement: 10 CFR 50.71(e) requires that the licensee periodically update the Final Safety Analysis Report to reflect the latest information developed. This includes information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirements since the original Final Safety Analysis Report or latest update. Contrary to this requirement, the licensee failed to update the Final Safety Analysis Report to reflect the correct minimum automatic depressurization system accumulator pressure value. Because this violation was of very low safety significance, was not repetitive or willful, and it was entered into the licensee's corrective action program as Condition Report CNS-2007-07468, this violation is being treated as a noncited violation, consistent with section VI.A.1 of the NRC Enforcement Policy. This violation is identified as NCV 05000298/2007011-06, Failure to comply with the requirements of 10CFR50.71.(e) and to assure the Updated Safety Analysis Report has the latest information developed.

4. OTHER ACTIVITIES

4OA5 Other Activities

b.1 Unresolved Item Associated with the Results of Modeling Of The Onset Of Vortexing Results Are Pending

Introduction: The team identified an unresolved item concerning the Emergency Condensate Storage Tanks (ECST) A/B volume design analysis. Specifically, the team identified that the licensee did not select an appropriate method for calculating the onset of vortexing during suction switchover from ECST to suppression pool. The licensee did not provide adequate technical justification for the methodology and available margin.

The team questioned the current condition and the licensee provided a corrective action evaluation, which provided a reasonable assurance that the current condition is not an immediate operability concern. Specifically, the licensee generated a list of expected scenarios and determined the amount of condensate volume used would not approach the swap-over setpoint. However, the license design basis specifically requires that the switchover volume be available and the emergency operating procedures do not allow the switchover to be over-ridden. The team expressed a concern over the lack of technical rigor, (i.e., non conservative methodology) and little available margin. The licensee generated Condition Report CNS-2007-07414 to address the team concerns.

Description: The team identified an unresolved item concerning the vortex calculation of the ECST and the available margins of the required volumes for switchover from ECST to suppression pool (torus) scenario and the 8-hour shutdown scenario. Specifically, the team identified that the licensee used an incorrect area for the calculation of the velocity and, therefore, the submergence Froude number, which would have an impact on the prediction to vortexing. Moreover, the available margins for the required volumes for switchover and 8-hour shutdown were minimal. The team reviewed Calculation NEDC 98-001, "Vortex Limit for the Emergency Condensate Storage Tanks A & B," Revision 1. The purpose of the calculation was to determine the minimum tank level required to prevent the formation of stable, air-core vortices during the switchover of high pressure coolant injection and reactor core isolation cooling from

ECST to suppression pool. The team noted that the ECST to suppression pool switchover scenario is described in Sub-Section 7.0, "Reactor Core Isolation Cooling System of Section IV, Reactor Coolant System of the USAR."

Calculation NEDC 98-001 stated that the required submergence of the ECST suction piping is based on Electric Power Research Institute Report TR-106266, "Inverted Draft Tubes to Improve Suction Performance of Vertical Pumps". A review of the formulas used in Calculation NEDC 98-001 and Electric Power Research Institute Report TR-106266 revealed that the statement in the calculation is incorrect. Electric Power Research Institute Report TR-106266 used pipe cross-sectional area in the calculation of the velocity (i.e., the same as used in NUREG/CR-2772, "Hydraulic Performance of Pump Suction Inlets for Emergency Core Cooling Systems in Boiling Water Reactors"). However, Calculation NEDC 98-001, incorrectly used the non conservative approach area, which is larger than the pipe cross sectional area, thus, resulting in lower velocity and, therefore, smaller Froude number.

The team reviewed Calculation NEDC 01-072, "ECCS [emergency core cooling system] Pump NPSH/Vortex Limit with Suction from CST [condensate storage tank]," Revision 0. This calculation determines the minimum condensate storage tanks water level required when the emergency core cooling system pump suctions are aligned to the CST during shutdown conditions. The team verified that, in contrast to Calculation NEDC 98-001, calculation NEDC 01-072 used the correct pipe cross sectional area for the calculation of the Froude number.

The incorrectly calculated Froude numbers in Calculation NEDC 98-001 were lower than 0.8, a cutoff limit for vortex creation derived from tests in NUREG/CR-2772 for a suction pipe equipped with strainer, and therefore vortexing was not identified in Calculation NEDC 98-001. However, approximate calculations performed by the team based on the ECST suction cross-sectional pipe cross-sectional area, rather than approach area, indicate that the use of approach area vs. pipe area results in a minimum submergence that is substantially lower than the minimum submergence calculated based on the pipe cross-sectional area. That is, vortexing could have occurred at a higher elevation submergence (i.e., earlier condensate depletion) had the pipe area been used. The review of the results of Calculation NEDC 98-001 indicated that a Froude number as high as 0.7934 was calculated (with 0.8 being the cutoff number for vortex creation).

Moreover, for a suction pipe with no strainer (as the case in Calculation NEDC 98-001), NUREG/CR-2772 recommends 0.7 as the cutoff Froude number. This suggests that vortexing would have occurred if this lower Froude number was used. The licensee justified the use of a larger cutoff Froude number of 0.8 because the suction from the condensate is an inverted 90-degree elbow, which should help retard vortex formation.

A second concern related to impact of the high pressure coolant injection and reactor core isolation cooling pump suction motor-operated valve stroke times on the switchover transient. During this transient there is a potential of air entrainment as the available volume is reduced to very low levels. The transient is largely affected by the assumed closing time of the high pressure coolant injection and reactor core isolation cooling pump suction motor-operated valves. The licensee used stroke-time results from in-service tests as part of the input to the design calculation. The stroke times used in Calculation NEDC 98-001 were chosen to conservatively bound the in-service test results. The calculation stated that "Within the limits of a hand-calculation, the results indicate very little margin exists to accommodate any unexpected increase in the stroke time of either high pressure coolant injection or reactor core isolation cooling pump suction valves. In fact the current operability limit for Motor-Operated Valve HPCI-MOV-MO58 should be lowered by 3 to 5 seconds." This is further discussed in Condition Report CNS-2006-09585, which states, "Note that the stroke time is within limits for the current NEDC 96-039 Revision 2, Status 1 (as-built) calc version. If the Revision 3, Status 2 "For Information Only" version of NEDC 96-039 were used, the vortexing calculation limits in NEDC 98-001 would need to be re-evaluated." Condition Report CNS-2006-09585 had been generated by the licensee in response to Information Notice 2006-21, "Operating Experience Regarding Entrainment of Air into Emergency Core Cooling and Containment Spray Systems."

With respect to current operability the team inquired whether the licensee had identified and evaluated the appropriate design basis scenarios for ECST drawdown to the low-level set-point for switchover of the high pressure safety injection/reactor core isolation cooling suction valves to the torus. The licensee generated Condition Report CNS-2007-07402 to address this issue and their evaluation demonstrated that there are no immediate operability concerns. The team reviewed Engineering Evaluation 01-090, Revision 1, which addressed condensate requirements for an 8-hour shutdown (Calculation NEDC 01-064, Revision 1) and station blackout (Calculation NEDC 89-1886, Revision 2). As part of the licensee operability evaluation, the licensee has stated emergency procedure bases provide guidance to maintain the ECST level between 60 and 120 inches by refilling from the condensate storage tank and that 60 inches minimum is above the ECST low level set-point (24.5 inches) for switchover to the torus but the Updates Safety Analysis Report does not allow the ECST to suppression pool (torus) switchover input to the emergency core cooling system to be over-ridden.

The team reviewed set-points in Calculation NEDC 92-050K, "HPCI-LS-074A/B and HPCI-LS- 75A/B Set points," Revision 2. The calculation used an analytical limit of 21.71 inches and refers to GE Calculation NEDC-32676P as its basis. Based on this analytical limit, Calculation NEDC 98-001 calculates the volume of the available water from both tanks is 9,918 gallons. The licensee takes credit of the approximately 400 gallons water in the ECST suction piping to meet the "at least 10,000 gallons" requirement stated in the Sub-Section 7.0, "Reactor Core Isolation Cooling System of Section IV, Reactor Coolant System of the USAR."

The minimum levels for the ECSTs were found in Calculation NEDC 01-064, Revision 1, references the requirement in Cooper Nuclear Station Procedure 6.LOG.601, Attachment 11, "ECST Level," Revision 90. The stated ECST minimum is 14.8 ft for each tank. Using 282 gallons per inch, a level of 14.8 ft equates to 50,083 gallons per tank or 100,166 gallons for the total from both tanks. The ECSTs consist of two separate tanks, which are cross-connected. The required volume of 86,072 gallons from both tanks is stated in Calculation NEDC 01-064, Revision 1. However, after depletion of 86,072 gallons the remaining volume is 14,094 gallons (i.e.,100,166 gallons minus 86,072 gallons) for two tanks, or 7,047 gallons for one tank, which corresponds to an

ECST tank level height of 25 inches from the bottom of each tank. This is only 0.5 inch over the switchover setpoint limit of 24.5 inches stated in Calculation NEDC 01-064. Based on this calculation, the licensee contends there will not be an unintended switchover to the torus, but the team was concerned about the minimal margin.

Given the low margin to switchover, the potential for air entrainment during this transient becomes important and must be considered. The team noted that NUREG/CR-2772 does not address submergence at the low end of the ECST drawdown when the suction valves are swapping over. Therefore, analytical evaluations of air entrainment in this region are not very reliable. Calculation NEDC 98-001 did not indicate any vortexing throughout the transient. Other sites, however, have seen test results indicating that vortexing would start at about 4.5 inches above the top of the pipe.

The licensee planned to perform a scaled test of the ECST switchover transient and assess the potential for vortex formation in the suction flow. Testing was in progress during the team inspection. Prior to leaving the site, the team was verbally informed of some preliminary results, which indicated a potential for air entrainment during the end of switchover transient. The team will review the modeling results to confirm assumptions on their operability and corrective actions and assess the impact on the remaining margin and any potential for margin erosion. This issue is unresolved item pending completion of this review. (URI) 05000298/2007011-07

b.2 Unresolved Item Regarding The Fuel Oil Storage Tank Required Submergence To Prevent Vortexing And Available Volume Are Marginal Without Accounting For Instrument Uncertainties

Introduction: The team identified an unresolved item regarding licensee application of instrument uncertainties and substantiating assumptions associated with the evaluation of vortexing and available volume in the emergency diesel generator fuel oil system.

Description: As a result of the team's questions regarding the available net positive suction head and vortexing potential in the diesel fuel oil systems, the licensee prepared Calculation NEDC 07-090, which was approved December 4, 2007 (subsequent to the team's departure from the site). The final results of Calculation NEDC 07-090 concluded that there was adequate available net positive suction head and vortexing would not occur in both the fuel oil storage and the day tanks. The team reviewed the calculation and did not identify any significant issues with respect to the calculation of net positive suction head since there was ample margin between the available and required net positive suction head. The team, however, found that the margin to avoid vortexing for the fuel oil storage tanks was approximately 0.323 inches. This is a concern with the team since this amount of margin is very low and may not have considered instrument uncertainties. The team determined this issue was an unresolved item.

The team requested the licensee to demonstrate how instrument uncertainty is accounted for in various engineering calculations and the basis for assumptions regarding the accounting for differences between water and fuel oil. Specifically, the calculations reviewed were:

- NEDC 07-090, "DG Fuel Transfer Pump Submergence Requirement", Revision 0

- NEDC 97-012, "Emergency Diesel Generator Fuel Oil On-Site Storage Technical Specification Requirements," Revision 2

- NEDC 87-052,"Emergency Diesel Generator Storage Tank Fuel Capacities," Revision 4

The description stated in Engineering Calculation 07-090,"DG Fuel Transfer Pump Submergence Requirements," Revision 0, is to determined the submergence requirements for the diesel generator fuel oil transfer pumps in the diesel generator fuel oil storage tanks and also determined the submergence requirements to prevent vortex formation at the diesel generator diesel oil pump suction inlets in the diesel generator diesel oil day tanks and to provide adequate inlet pressure to the diesel generator diesel oil engine driven and booster fuel pumps.

Calculation NEDC 07-090 concluded that the available net positive suction head was adequate and vortexing would not occur in the diesel generator fuel oil storage tanks or fuel oil day tanks. The findings suggest there are no issues with the net positive suction head results for fuel tanks, since there are several feet of margin available in both. The team however, had the following concerns regarding minimum margin:

a) In regards to fuel oil storage tank inventory, vortexing may have a more significant impact since there is a minimal margin for the fuel oil storage tanks between the 8 feet-4.0 inches or 49,768 gallons. For a 7-day run technical specifications require 49,500 gallons, this is only a margin of 268 gallons.

b) Part of the calculation conclusion indicated that the submergence criteria used in the calculation to prevent vortex formation are based on Froude number correlations developed from testing performed with water instead of diesel fuel. They also state that submergence required to prevent vortex formation with diesel fuel will be higher than water and, therefore, a 20 percent allowance was added to the critical submergence values calculated. The conclusion in the Calculation NEDC 07-090 states, "The critical submergence values calculated herein would still remain below the actual submergence at the minimum tank level even with this conservative 20 percent factor added." This assumption of a 20 percent allowance for difference between water and fuel was not substantiated for the fuel oil tanks. Given this assumption, the margin to vortexing is reduced from 1.54 to 0.323 inches.

c) Since margins are minimal, instrument uncertainty considerations become very important and it is not obvious how instrument uncertainties are considered in the licensee's fuel oil calculations. Specifically, in the conclusions found in Calculation NEDC 07-090 and in the revision to Calculation NEDC 87-052 . The team identified that Revision 4 to Calculation NEDC 87-052, which is used to calculate fuel oil tank capacities, deleted the reference to Calculation NEDC 91-198, "Diesel Generator Fuel Oil Tank Level instruments Accuracy," Revision 0. The reason given for this deletion was not clear.

d) Discrepancies between calculations that evaluate fuel oil storage tank volumes. Specifically, considering the fuel storage tanks, Calculation NEDC 97-012, "Emergency Diesel Generator Fuel Oil On-site Storage Technical Specification Requirements," concluded that the minimum administrative limit based on 8 feet-4.0 inches plus 8.5 inches (for instrument uncertainty) is 9 feet 0-1/2 inches. However Calculation NEDC 87-052," Emergency Diesel Generator Storage Tank Fuel Capacities," Revision 4, stated that at 8 feet-4 inches the calculated volume is 49,768 gallons. The required volume for a 7-day run is 49,500 gallons.

Following the telephone exit on December 12, 2007, the licensee identified two additional issues concerning the diesel fuel oil day tanks and the storage tanks, which are documented in Condition Reports CNS-2007-08590 and CNS-2007-8682, respectively. These issues are related to the licensee's failure to account for vortexing impact on available fuel oil volume, and not considering the impact of instrument uncertainties on measuring the fuel oil storage tank volumes.

Additional analysis is needed from the licensee to determine whether instrument uncertainly has been adequately accounted for and if adequate margin exists to ensure usable volume remains above the minimum submergence limits in the emergency diesel generator system fuel oil tanks. Therefore, this issue is considered as an unresolved item pending additional information from the licensee and a final review of this analysis by the NRC staff. The licensee has written Condition Report CNS-2007-08482 to address the team's questions regarding Calculation NEDC 07-09.
(URI 05000298/2007011-08)

4OA6 Meetings, Including Exit

Exit Meeting Summary

On November 2, 2007, the team presented the baseline inspection results to Paul Fleming, Director of Nuclear Safety Assurance, and other members of Cooper Nuclear Station staff at the conclusion of the onsite inspection. The licensee acknowledged the findings presented. The team verified that proprietary information which was reviewed was returned. On December 12, 2007, a telephone conference was held to present to the licensee staff changes to the initial characterization of one finding and to discuss two unresolved items. A telephone discussion was held on January 22, 2008, to clarify the characterization of a finding and request status of an engineering evaluation.

4OA7 Licensee-Identified Violations

No significant findings were identified

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

Jim Flaherty, Senior Staff Licensing Engineer
Scott S. Freborg, Valve Group Supervisor-ESD
Michael T. Boyce, Director of Projects
Michael J. Colomb, General Manager of Plant Operations
Paul Fleming, Director of Nuclear Safety Assurance
Gary Kline, Director of Engineering
Vasant Bhardwaj, Manager- ESD
Daniel Buman, System Engineering- Manager
Roman Estrada, Manager-Corrective Action
Todd Stevens, Manager-Design Engineering
David VanDerKamp, Manager-Licensing
Dave Werner, Operations Training- Supervisor
Mark Bergmeier, Operations Support Group- Supervisor
Stan Domikaitis, Mechanical Design- Supervisor
Gabe Gardner, Civil Design Supervisor-Design Engineering
Marshall VanWinkle, Electrical Supervisor-Design Engineering
Eric Nelson, Electrical Design Engineer
Raymond Rexroad, Electrical System Engineer
Jeff Ehlers, System Engineer-SED
Ole Olson, ESD - Risk Management
Gerald Horn, Engineering Specialist-Design Engineering
Kenneth Done, Senior Staff Engineer
Edward Holcomb, Mechanical Engineer-Design Engineering
George Levy, Mechanical Engineer
Mark Unruh, Senior Staff Engineer
Mark F. Metzger, System Engineer-SED

NRC personnel

Kenneth Heck, Quality & Vendor Branch, DCIP/NRO
Nick Taylor, Senior Resident Inspector

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

05000298/2007011-01	NCV	Failure to consider vortexing and available net positive suction head impact on the emergency diesel fuel oil system. (Section 1R21.b.1)
05000298/2007011-02	NCV	Installation of essential electrical cable with inadequate fault current ratings and not in accordance with original design basis requirements. (Section 1R21.b.2)

05000298/2007011-03	NCV	Failure to ensure that design bases information remain consistent within affected design documents. (Section 1R21.b.3)
05000298/2007011-04	NCV	For failure to comply with design control program requirements. (Section 1R21.b.4)
05000298/2007011-05	NCV	Inadequate corrective actions associated with multiple workmanship issues on safety-related valves. (Section 1R21.b.5)
05000298/2007011-06	NCV	Severity Level IV for failure to comply with the requirements of 10 CFR 50.71(e) and to assure the Updated Safety Analysis Report has the latest information developed. (Section 1R21.b.6)
05000298/2007011-07	URI	Results of modeling of the onset of vortexing results are pending. (Section 4AO5.b.1)
05000298/2007011-08	URI	Fuel oil storage tank required submergence to prevent vortexing and available volume are marginal without accounting for instrument uncertainties. (Section 4AO5.b.2)

LIST OF DOCUMENTS REVIEWED

Calculations

Number	Title / Description	Revision/Date
NEDC 96-039	DC Powered MOV Stroke Time and Capability	1; 1C1; Calculation 2; and 3
NEDC 98-001	Vortex Limit For The Emergency Condensate Storage Tanks A & B	1
NEDC 00-110	MOV Program Valve Margin Determination	3
NEDC 91-078	System Level Design Basis Review of High Pressure Coolant Injection (HPCI) System Program MOVs	3
NEDC 91-080	System Level Design Basis Review of Residual Heat Removal (RHR) System Program MOVs	6
NPP0011-CALC-001	Methodology Developed By MPR in Conjunction With BWROG NEDC-32958	0

Number	Title / Description	Revision/Date
NEDC 95-003	Determination of Allowable Operating Parameters for CNS MOV Program MOVs	17
NEDC 86-105C	DC Short Circuit Study	4
NEDC 86-105D	Critical DC Bus Coordination Study	7
NEDC 87-131B	250 Volt Division II Load and Voltage Study	8
NEDC 87-131C	125 Volt Division I Load and Voltage Study	9
NEDC 91-044	Cable Resistance Calculation for 125 and 250 VDC Loads	4
NEDC 91-094	125VDC / 250 VDC Battery Charger Analysis	5
NEDC 91-176	DC Systems High Voltage	2
NEDC 91-185	MOV Thermal Overload Heater Sizing	2
NEDC 93-022	NED Review of Erin MOV Calc. C122-89-10.039	5
NEDC 00-042	AC MOV Voltage Drop Calculation	1
NEDC 03-028	Component Level Calculation PC-AOV-245, 2797A&B	11
NEDC 05-013	AOV Component Level Calculation PC-AOV-237AV	1
EEN-05-017	Evaluate Operability Requirements for ESST	0
NPP1-PR-01	Station Blackout Coping Assessment [Enercon Services]	2
NEDC 92-050K	HPCI-LS-74A/B and HPCI-LS-75 A/B ECST Setpoints	2/May 26, 1998
NEDC 01-072	ECCS Pump NPSH/Vortex Limit with Suction from CST	0/January 17, 2002
NEDC 01-064	8-hour ECST Volume Requirements for an Isolated Reactor	1/August 22, 2007
NEDC 89-1886	CNS Station Blackout (SBO) Condensate Inventory	2/August 22, 2007
NEDC 93-184	RHR Heat Exchanger Thermal Performance and Tube Plugging Margin	1/October 7, 2002
NEDC 91-239	DGLO/DGJW/DG Intercooler Heat Exchanger Evaluation	2/September 7, 2007

Number	Title / Description	Revision/Date
NEDC 07-090	DG Fuel Transfer Pump Submergence Requirements	0
NEDC 94-142	Core Spray Flows with Minimum Flow Valve Open	3/November 30, 1998
NEDC 97-012	Emergency Diesel Generator Fuel Oil On-Site Storage Technical Specification Requirements	2
NEDC 87-052	Emergency Diesel Generator Storage Tank Fuel Capacities	4
NEDC 88-086B	Setpoint Determination of Second Level Undervoltage Relays	10/July 20, 2006
NEDC 91-157	DG Transient Analysis	1/February 8, 1995
NEDC 00-003	CNS Aux. Power System Load Flow and Voltage Analysis	5/November 29, 2006
NEDC 00-111	CNS Auxiliary Power System AC Loads	4/November 29, 2005
NEDC 91-190	AC Equipment and Cable Short Circuit Withstand Ratings	1/February 16, 1993
NEDC 86-105E	AC Bus Short Circuit Study	3/January 28, 1993
NEDC 86-105B	CNS Critical AC Bus Coordination Study	7/September 11, 2001
NEDC 93-104	Emergency Transformer Permissive Relay Setpoint Calculation	3/July 26, 2006
NEDC 88-086B	Setpoint Determination of Second Level Undervoltage Relays	10/July 26, 2006
NEDC 91-208	Review of B&R Calc. 2.09.06 sheets 20 – 22B	0/July 9, 1991
NEDC 92-204	NED Reviews of Seismic Qualification of Potter-Brumfield Relays & IPS Converter	February 8, 1993
NEDC 88-306	Analysis of ADS Accumulators	0/January 10, 1989

Number	Title / Description	Revision/Date
2.05.06	Burns and Roe - Calculations for Minimum Cable Size – 4160 V	May 14, 1970
NEDC 88-086B	Setpoint Determination of Second Level Undervoltage Relays	10/July 20, 2006

Design Criteria Documents

Number	Title / Description	Revision/Date
DCD-04	AC Electrical Distribution Systems	4/11/05
DCD-05	DC Electrical Distribution Systems	6/24/04
DCD-12	Core Spray	9/13/02
DCD-01	Diesel Generator	5/31/05
PBD-EQ	Program Basis Document-Environmental qualification. Volume 1	4
DCD-2	High Pressure Coolant Injection	12/23/04

Training Documents

Number	Title / Description	Revision/Date
Scenario 1	CDBI scenario 1, LOOP stuck open (cycling) SRV	00
Scenario 2	CDBI scenario, SBO small break LOCA	00
EP 5.3EMPWR	Emergency Power	5/25/07
EP 5.3SBO	Station Blackout	7/10/07
EP 5.3Alt-Strategy	Alternative core cooling mitigating strategies	9/27/07
5.3Alt-strategy, 200383A0501	RCIC Manual Operation 5.3Alt-strategy,	01
SKL034-40-85	Inject Fire Protection Water to RCIC	00
364036I0102	Isolation, startup and loading of Diesel Generator	14

Number	Title / Description	Revision/Date
EP 5.4Fire-SD	Fire induced shutdown from outside control room	7/13/07
EP 5.3SBO	Station Blackout	7/10/07
EP 5.8.6	RPV Flooding Systems (Table 6)	10/15/07
EP 5.8.2	Alternate emergency Depressurization (Table 2)	7/18/07
EP 5.8.7	Primary Containment Flooding/Spray Systems	10/18/07
EP 5.8.20	EOP Plant Temporary Modifications	7/23/03
EP 5.8.13	Outside Shroud Injection Systems (Failure to Scram) Table 13	5/25/04

Procedures

Number	Title / Description	Revision/Date
0.29.2	USAR Control and Maintenance	15
3.1	Engineering Definitions	7
3.32	Design Basis Program Description	6
3.32.4	Logic Diagram Development	1
3.32.5	Configuration Matrix Development	1
3.32.7	Document review	5
3.32.8	DCD Verification and Validation	1
3.32.9	Design Criteria Document Records Control	4
3.32.11	Design Criteria Document Punch List	1
3.33	Motor Operated Valve Program	18
3.4.7	Design Calculations	28
2.2.71	Service Water System	97
2.2.33.1	High Pressure Coolant Injection System Operations	26

Number	Title / Description	Revision/Date
EDP-06	Temporary Configuration Change	21
EDP-16	Design Criteria Document Production and Control	03
7.3.8.2	Diesel Generator Electrical Examination and Maintenance	20/ 08/13/2007
6.1DG.302	Surveillance Procedures, Undervoltage Logic Functional & Load Shedding & Sequential Loading Test (Div 1	38/ 10/28/2006
6.2DG.302	Undervoltage Logic Functional, Load Shedding, and Sequential Loading Test (Div 2)	33/ 11/09/2006
6.1EE.303	Emergency Bus Undervoltage (27) Relays Testing and Calibration (Div 1)	7/ 11/14/2005
6.EE.301	Emergency Bus Undervoltage Relays Testing and Calibration	6/ 11/24/2004
6.1EE.302	4160V Bus 1F Undervoltage Relay and Relay Timer Functional Test (Div 1)	17/ 6/15/2007
2.2.20.1	Diesel Generator Operations	38/ 10/03/2007
6.2DG.102	Diesel Generator Demonstration of Operability Test (DIV 2)	31/ 8/15/2007
6.2DG.101	Diesel Generator 31-Day Operability Test (IST) (DIV 2)	49/ 10/2/2007
6.ADS.302	ADS Accumulator Functional Test	8/ 12/7/2006
3.4	Configuration Change Control	45/ 7/31/2007
EDP-06	Supporting Requirements For Configuration Change Control	21/ 9/24/2007
3.15	Procurement Document Review	12/ 2/8/2007
5.8.20	Emergency Operating Procedure, EOP plant Temporary Modifications	7/23/03
5.8	Emergency Procedure, attachment 1	13
BWROG EPGs/SAGs	BWR Owners' Group Emergency Procedure and Server Accident Guidelines, Appendix D: Comparison to Revision 4 EPGs	2
BWROG EPGs/SAGs	BWR Owners' Group Emergency Procedure and Severe Accident Guidelines, Appendix B: Technical Basis, Vol 1	2
6.EE.607	125V Station Battery Performance Discharge Test	13
6.EE.609	125V/250V Battery Intercell Connection Testing	11

Number	Title / Description	Revision/Date
6.1RPS.301	Manual Scram Functional Test (Div 1)	3
6.2RPS.301	Manual Scram Functional Test (Div 2)	3
7.3.26	Raychem WCSF-N Insulated Splices (Bolted)	8
7.3.26.10	EGS Grayboot Installation	3
7.3.27.1	125V Station Battery Equalizing Charge	1
7.3.28.1	EQ and Essential Lead Removal/Installation	6
STP 92-034	DC Motor Performance Test -	07/02/92

Surveillance Procedures

Number	Title / Description	Revision
6. RCIC.201	RCIC Power Operated Valve Operability Test (IST)	14
6.2 RHR.101	RHR Test Mode Surveillance Operation (IST)(Div 1)	19
6.2 RHR.101	RHR Test Mode Surveillance Operation (IST)(Div 2)	21
6. HPCI.103	HPCI IST and 92 Day Test Mode Surveillance Operation	33
6. HPCI.201	HPCI Valve Operability Test (IST)	14
6. RCIC.102	RCIC IST and 92 Day Test	22

Drawings

Number	Title/Description	Revision/Date
3012, Sh 3	Cooper Nuclear Station Main Three Line Diagram	N17/ 09/28/2004
3012, Sh 5	Main Three Line Diagram	N146/14/2005
14EK-0144	Cooper Nuclear Station Contract E69-21, Diesel Engine Generator Schematic Diagram	N17/ 12/28/2006
14DK-0842	Schematic Voltage Regulator	N03/ 06/25/2007
14DK-0843	Cooper Nuclear Station Schematic & Interconnection Diagram	N04/

Number	Title/Description	Revision/Date
	Series Booster Exciter Voltage Regulator	06/25/2007
14DK-0844	Schematic & Interconnection Diagram For Motor Operated Potentiometer	N01/ 06/25/2007
3070	Electrical Symbol List, Cooper Nuclear Station	N06/ 12/30/2005
3032, Sh 1	Cooper Nuclear Station Control Elementary Diagrams,	N04/ 8/17/1999
3004, Sh 3	Cooper Nuclear Station Auxiliary One Line Diagram, MCC C, D, H, J, DG! & DG2	N20/ 12/1/1999
3012	Cooper Nuclear Station Main Three Line Diagram Sheet #3	N17/ 09/28/2004
2031 Sh 2	Reactor Building – Closed Cooling Water System	N64
2036 Sh 1	Reactor Building Service water System	N93
2006 Sh 1	Circulating Screen Wash & Service Water Systems	N70
2006 Sh 2	Circulating Screen Wash & Service Water Systems	41
2006 Sh 3	Circulating Screen Wash & Service Water Systems	52
2006 Sh 4	Control Building Service Water System	N45
2040 Sh 1	Residual Heat removal System	N76
2040 Sh 2	Residual Heat removal SYS Loop ‘B’	N15
2041	Reactor Building Main Steam System	N79
2043	Reactor Core Isolation Coolant and Reactor feed Systems	N50
2044	High Pressure Coolant Injection and Reactor Feed Systems	69
2045 Sh 1	Core Spray System	N54
2022	Primary Containment Cooling & Nitrogen Inerting	78
3001	Main One Line Diagram	N15
3002, Sh 1	Auxiliary One Line Diagram, Swgr.	N43
3003, Sh 2	Auxiliary One Line Diagram, MCCs	N41
3004, Sh 3	Auxiliary One Line Diagram, MCCs	N20
3005, Sh 5	Auxiliary One Line Diagram, MCCs	N48
3006, Sh 5	Auxiliary One Line Diagram, MCCs & Starter Racks	N72

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3007, Sh 6	Auxiliary One Line Diagram, MCCs	N80
3009, Sh 1	12.5 KV Bus System One Line Diagram	N29
3010, Sh 1	Vital One Line Diagram, Sheet #3	N17
3012	Main Three Line Diagram	N10
3016, Sh 11	4160V Switchgear Elementary Diagram	N09
3017, Sh 1	4160V Switchgear Elementary Diagram	N10
3018, Sh 2	4160V Switchgear Elementary Diagram	N09
3019, Sh 3	4160V Switchgear Elementary Diagram	N29
3020, Sh 4	4160V Switchgear Elementary Diagram	N19
3035, Sh 4	Control Elementary Diagram - Containment	N31
3037, Sh 4A	Control Elementary Diagram - Containment	N01
3037, Sh 6	Control Elementary Diagram	N32
3040, Sh 9	Control Elementary Diagram	N31
3045, Sh 14	Control Elementary Diagram	N44
3045, Sh 16	Control Elementary Diagram	N12
3067, Sh 19	Control Elementary Diagram	N33
3058	DC One Line Diagram	N47
3059, Sh 1	DC Panel Schedules	N34
3059, Sh 11	125VDC Load & Fuse Schedule	N06
3059, Sh 12	125VDC Load & Fuse Schedule	N09
791E261, Sh 1	Residual Heat Removal System [Schematics]	N16
791E261, Sh 2	Residual Heat Removal System [Schematics]	N13
791E261, Sh 4	Residual Heat Removal System [Schematics]	N16
791E261, Sh 5	Residual Heat Removal System [Schematics]	N18
791E261, Sh 7	Residual Heat Removal System [Schematics]	N16

Number	Title/Description	Revision/Date
791E261, Sh 8	Residual Heat Removal System [Schematics]	N20
791E261,Sh13	RHR System Schematic for MOV TOL Devices	N08
791E261,Sh15	RHR System [Schematic for MOV 10-89A]	N11
791E261 Sh16	RHR System [Schematic for MOV 10-39A]	N08
791E261 Sh19	RHR System [Schematic for MOV 10-89B]	N22
791E261,Sh 20	RHR System [Schematic for MOV 10-39A]	N13
791E264, Sh 1	RCIC System [Schematics]	N12
791E264, Sh 2	RCIC System [Schematics]	N25
791E271,Sh 1	HPCI System [Schematics]	N47
791E271 Sh1A	HPCI System [Schematics]	N06
791E271, Sh 2	HPCI System [Schematics]	N18
791E271, Sh 3	HPCI System [Schematics]	N21
791E271, Sh 4	HPCI System [Schematics]	N24
791E271,Sh4A	HPCI System [Schematics]	N05
791E271, Sh 5	HPCI System [Schematics]	N23
791E271, Sh 6	HPCI System [Schematics]	N19
791E271,Sh6A	HPCI System [Schematics]	N04
791E271, Sh 7	HPCI System Valves 23-14 & 19 [Schematics]	N19
791E271, Sh 8	HPCI System Valve 23-25 [Schematics]	N19
791E271, Sh 9	HPCI System [Schematics]	N19
791E271,Sh10	HPCI System [Schematics]	N20
791E271 Sh11	HPCI System [Schematics]	N00
E507, Sh.39	Reactor Building Connection Diagram	N02
EQ-111, Sh1	EQ Configuration Detail - ASCO SOVs	N02

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EQ-111, Sh2	EQ Configuration Detail - ASCO SOVs Tabulation	N02
EQ-116, Sh1	EQ Configuration Detail - NAMCO Limit Switches	N03
EQ-116, Sh1	EQ Configuration Detail - NAMCO Limit Switches	N01
G5-262-743, Sh.1	EDG #1 Electrical Schematic	N21
G5-262-743, Sh.1A	EDG #1 Electrical Schematic	N04
G5-262-743, Sh.10	EDG #2 Electrical Schematic	N13
KSV-47-8	EDG 1&2 Cooling Water Schematic	N24
NB64032	69-12.5 KV One Line Diagram	7
NC29546	Transmission Line Routes	3

<u>Miscellaneous</u>

Number	Title / Description	Revision/Date
PSA-ES019, USAR	CNS Risk Information Matrix	0
Design Change 93-024	Diesel Generator Upgrades	12/06/1995
CED 6022440	Change Evaluation Document Surge Suppressor Replacement in DG Controls	10/18/2006
Root Cause Investigation Report CR-CNS-2005-1360	Root Cause Investigation Report ; DG1 Control Power Failure and Loss of Shutdown Cooling During Sequential Load Test	03/07/2005
Laboratory Report	Southwest Research Institute Component Analysis Failure Analysis of Transient-Suppression Network	02/22/2005
CR-CNS-2005-8336	Apparent Cause Evaluation Large DG kVAR spikes	12/15/2005
Report EDG	Diesel Generator Voltage Droop Not Bypassed for Emergency Signal Presentation to Corrective Action Review Board	2/ 06/1/2006
Appendix A Logic Diagram	Design Criteria Document, AC Electrical Distribution, Appendix A Medium Voltage Breaker Alignment Handout	10/22/2007
CR-CNS-2006-05357	Essential control relay contacts are not rated for their application in the DG control circuits.	8/29/2006

Number	Title / Description	Revision/Date
EGS-TR-23078-9013-02	Test Report for Relay Cycle Aging Test	3/28/2007
10394-A004/NUPIC 19273	NUPIC Audit of MPR Associates Inc. Quality Assurance Program	9/16/2005
10394-A004/NUPIC 19273	Revision to the Engineering PBSA Worksheet, Rev 1	3/07/2006
10394-A004/NUPIC 19273	Closeout of NUPIC Audit of MPR Associates Inc. Quality Assurance Program	3/16/2006
NEDC-31366 Ltr.	Letter from B. Boger to R. Pinelli Revision to Safety Evaluation Report on NEDC-31366, Instrument Setpoint Methodology (NEDC-31336P)	11/6/1995
Email 01	Email between Eric M. Nelson, CNS and Bob Ryan, Okonite Company Cable Withstand Short Circuit Currents	10/19/2007
NLS9100224	EDS Design Licensing Commitments	4/8/1991
IEEE 308-1970	IEEE Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations	1970
ANSI/IEEE 313-1971	IEEE Standard for Relays and Relay Systems Associated with Electric Power Apparatus	1971
IEEE Std 279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations	1971
ICEA S-66-524, NEMA WC 7	Cross-Linked-Polyethylene-Insulated Cable	7
EDE-38-1090, Revision 0 (Proprietary)	Setpoint Calculation Guidelines for the Cooper Nuclear Station	1/25/1991
NEDC-31336P-A (Proprietary)	General Electric Instrument Setpoint Methodology	September 1996
MP 93-003	LOGT & TGF Modification (Salem Event)	7/22/2003
ESC 91-152	Kurman Relay and Motorola Power Supply Replacement in Diesel Generators 1 & 2	4/26/2003
SKL060-35-14	Plant Modification Change Evaluation Document Practical	3
SKL060-35-15	Nuclear Experience for CEDs	0
ESP00006052	3.4 Process Modification Classroom Course	4
OER Document: IN 95-	Undervoltage Protection Relay Settings Out of	4/18/1995

Number	Title / Description	Revision/Date
05	Tolerance due to Test Equipment Harmonics	
OE Misc.	List of Operating Experience associated Condition Reports	October 3, 2007
CNSLO 2005-00021	Motor Operated Valve Program Focused Assessment Report	August 15-19, 2005
ERN 10548876	Engineering Request Notification	0
PSA-ES019	CNS Risk Information Matrix	0
INT008-06-09	Operation Lesson Plan, RPV control, RPV Level	18
EOP CN 5.8.13	EOP Change Notice, Outside Shroud Injection Systems (Failure to Scram) Table 13	June 8, 1993
CED 6006512	Change Evaluation Document for Cable DC71(P) Extension for HPIC-MO-19	12/02/01
EE 03-007	Engineering Evaluation, Review of SLR (System Level Review) Calculations and Incorporation into NEDC 95-003	02/19/03

Vendor Documents

Number	Title / Description	Revision/Date
VM-0246	Series Booster Exciter Regulator	9/21/2007
IB 7.4.1.7-7, Issue E	Instructions, Single Phase Voltage Relays	E
ICEA P-32-382	Short Circuit Characteristics of Insulated Cables	3/1969
Work Order 2520-02 Burns and Roe, Inc.	Calculations of Minimum Cable Size – 4160V	5/14/1970
Vendor Manual	Potter & Brumfield Datasheet for KRPA, KRP, KA, KR series relays	0
VM-1188	C&D Charter Power System Inc, 125 & 250 Volt Batteries and Chargers	7/ 3/30/1999
Vendor Manual - E7000	AGASTAT Nuclear Qualified Time Delay Relays – Series E7000	4/24/2002
Vendor Manual -EGP	AGASTAT Nuclear Qualified Control Relays – Series EGP/EML/ETR	4/24/2002

CD 7.4.1.7-7, Issue A	ITE-27N/ITE-59N Undervoltage/ Over-voltage Relays	1
11983	Instruction Manual for Protective Relay Test Set Model EPOCH-10	2/ 1992
VM-0394	Rubber Seat Butterfly Valves H&V	19
VM-0988	Allis-Chalmers Rubber Seat Butterfly Valves Fisher Controls Composite Manual	50
VM-0277	Cast Steel Valves	68

<u>Surveillance and Inservice Tests</u>

Number	Title / Description	Revision/Date
HPCI-MOV-MO14	Inservice Test Results	8/20/03, 5/26/04, 3/2/05, 12/7/05, 9/13/06, 6/20/07, 9/20/07
HPCI-MOV-MO19	Inservice Test Results	8/19/03, 6/13/04, 4/7/05, 1/31/06, 11/25/06, 9/20/07
HPCI-MOV-MO25	Inservice Test Results	8/19/03, 6/13/04, 4/7/05, 1/31/06, 11/25/06, 9/20/07
RCIC-MOV-MO18	Inservice Test Results	9/6/07
RCIC-MOV-MO41	Inservice Test Results	9/6/07
RHR-MOV-MO39A	Inservice Test Results	7/16/03, 4/30/04, 2/14/05, 11/30/05, 9/16/06, 7/3/07
RHR-MOV-MO39B	Inservice Test Results	6/15/03, 4/10/04, 2/4/05, 12/1/05, 9/27/06, 7/24/07
SW-MOV-MO89A	Inservice Test Results	4/6/00, 7/30/01, 11/22/02, 3/16/04, 7/9/05, 11/1/06
SW-MOV-MO89B	Inservice Test Results	3/22/00, 7/20/01, 11/17/02, 3/17/04, 7/15/05, 11/12/06
HPCI Pump	Inservice Test Results	9/20/07
RCIC Pump	Inservice Test Results	9/5-6/07
Division1RHR Pumps1A/1C	Inservice Test Results	7/3/07

Condition Reports

CR-CNS-2004-03375	CR-CNS-2006-05357	CR-CNS-2007-06619
CR-CNS-2004-03867	CR-CNS-2006-06025	CR-CNS-2007-06620
CR-CNS-2005-00095	CR-CNS-2006-06797	CR-CNS-2007-06697
CR-CNS-2005-00811	CR-CNS-2006-07083	CR-CNS-2007-06749
CR-CNS-2005-00919	CR-CNS-2006-07401	CR-CNS-2007-06778
CR-CNS-2005-01013	CR-CNS-2006-07673	CR-CNS-2007-06792
CR-CNS-2005-01208	CR-CNS-2006-08706	CR-CNS-2007-06836
CR-CNS-2005-01360	CR-CNS-2006-09057	CR-CNS-2007-06838
CR-CNS-2005-01367	CR-CNS-2006-09096	CR-CNS-2007-06855
CR-CNS-2005-01606	CR-CNS-2006-09110	CR-CNS-2007-06863
CR-CNS-2005-01805	CR-CNS-2006-09166	CR-CNS-2007-06882
CR-CNS-2005-01833	CR-CNS-2006-09493	CR-CNS-2007-06933
CR-CNS-2005-02317	CR-CNS-2006-09932	CR-CNS-2007-07333
CR-CNS-2005-02894	CR-CNS-2006-09933	CR-CNS-2007-07339
CR-CNS-2005-02981	CR-CNS-2007-00027	CR-CNS-2007-07340
CR-CNS-2005-05162	CR-CNS-2007-00164	CR-CNS-2007-07361
CR-CNS-2005-05287	CR-CNS-2007-00653	CR-CNS-2007-07361
CR-CNS-2005-06210	CR-CNS-2007-01113	CR-CNS-2007-07385
CR-CNS-2005-08103	CR-CNS-2007-01454	CR-CNS-2007-07386
CR-CNS-2005-08336	CR-CNS-2007-01812	CR-CNS-2007-07402
CR-CNS-2005-08375	CR-CNS-2007-02127	CR-CNS-2007-07409
CR-CNS-2005-08641	CR-CNS-2007-02699	CR-CNS-2007-07414
CR-CNS-2005-08695	CR-CNS-2007-02818	CR-CNS-2007-07421
CR-CNS-2005-08769	CR-CNS-2007-03860	CR-CNS-2007-07423
CR-CNS-2005-08771	CR-CNS-2007-03974	CR-CNS-2007-07434
CR-CNS-2005-08865	CR-CNS-2007-04339	CR-CNS-2007-07459
CR-CNS-2005-09011	CR-CNS-2007-05395	CR-CNS-2007-07461
CR-CNS-2005-09642	CR-CNS-2007-05571	CR-CNS-2007-07468
CR-CNS-2006-00554	CR-CNS-2007-05582	CR-CNS-2007-07478
CR-CNS-2006-00869	CR-CNS-2007-05682	CR-CNS-2007-07571
CR-CNS-2006-02097	CR-CNS-2007-05769	CR-CNS-2007-07572
CR-CNS-2006-02169	CR-CNS-2007-06143	CR-CNS-2007-07585
CR-CNS-2006-03083	CR-CNS-2007-06492	CR-CNS-2007-07608
CR-CNS-2006-03093	CR-CNS-2007-06550	CR-CNS-2007-07609
CR-CNS-2006-03562	CR-CNS-2007-06615	CR-CNS-2007-08482

Work Orders

4478595	4439530	4308827
4488858	4499506	00-1721
4473999	4499507	00-1765