UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
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ARLINGTON, TEXAS 76011-4125

March 17, 2011

EA 11-024

Brian J. O'Grady, Vice President-Nuclear
and Chief Nuclear Officer
Nebraska Public Power District
Cooper Nuclear Station
72676 648A Avenue
Brownville, NE 68321

SUBJECT: COOPER NUCLEAR STATION - NRC TRIENNIAL FIRE PROTECTION INSPECTION REPORT 05000298/2010006; PRELIMINARY WHITE FINDING

Dear Mr. O'Grady:

On November 5, 2010, the U.S. Nuclear Regulatory Commission (NRC) completed an inspection at the Cooper Nuclear Station. The enclosed inspection report documents the inspection results, which were discussed in an exit meeting on March 14, 2011, with Mr. D. Buman, Director of Engineering, and other members of your staff.

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

Based on the results of this inspection, the NRC has identified two findings that were evaluated for risk under the Significance Determination Process. Violations were associated with each of the findings.

The attached report discusses a finding that was preliminarily determined to be a White finding, a finding with low-to-moderate increased safety significance which may require additional NRC inspections. This finding was assessed based on the best available information, including influential assumptions, using the applicable Significance Determination Process (SDP). As described in Section 1R05.01 of the attached report, this finding involves the failure to verify that procedure steps to safely shutdown the plant in the event of a fire would actually reposition three motor operated valves to the required positions and the concurrent failure to address a previous finding that involved the same procedure steps. This finding has preliminary low-to-moderate safety significance because it involves multiple fire areas and risk factors that were not dependent on specific fire damage. The scenarios of concern involve larger fires in specific areas of the plant which trigger operators to implement fire response procedures to place the plant in a safe shutdown condition. Since performing some of those actions using the procedures as written would not have aligned three valves to their required positions, this would challenge the operators' ability to establish adequate core cooling. This finding does not represent an immediate safety concern because your staff promptly changed the procedures to locally reposition position the valves.

This finding is also an apparent violation of NRC requirements and is being considered for escalated enforcement action in accordance with the NRC Enforcement Policy. The current Enforcement Policy is included on the NRC's web site at http://www.nrc.gov/about-nrc/regulatory/ enforcement/enforce-pol.html.

In accordance with Inspection Manual Chapter 0609, we intend to complete our evaluation using the best available information and issue our final determination of safety significance within 90 days of this letter. The significance determination process encourages an open dialog between the staff and the licensee; however the dialogue should not impact the timeliness of the staff's final determination. Before we make a final decision on this matter, we will hold a Regulatory Conference to provide you an opportunity to present to the NRC your perspectives on the facts and assumptions used by the NRC to arrive at the finding and assess its significance. The Regulatory Conference should be held within 30 days of the receipt of this letter and we encourage you to submit supporting documentation at least one week prior to the conference in an effort to make the conference more efficient and effective. This Regulatory Conference will be open for public observation.

At the Regulatory Conference, in addition to providing your perspectives on the finding and the significance, please be prepared to discuss (1) the cause(s) for the performance deficiency, (2) corrective actions taken or planned for the performance deficiency, and (3) the reasons why your corrective actions for Violation 05000298/2008008-01, a finding with low-to-moderate safety significance, were not adequate to verify that the procedure would have worked as intended.

Please contact Neil O'Keefe at (817) 860-8137 within 10 days of receipt of this letter to schedule a date for the Regulatory Conference. If we have not heard from you within 10 days, we will continue with our significance determination and enforcement decision. The final resolution of this matter will be conveyed in separate correspondence.

Because the NRC has not made a final determination for this matter, no Notice of Violation is being issued for this inspection finding at this time. In addition, please be advised that the characterization of the apparent violation described in the enclosed inspection report may change as a result of further NRC review.

Based on the results of this inspection, the NRC has also identified one additional issue that was evaluated under the risk significance determination process as having very low safety significance (Green). The finding was determined to involve a violation of NRC requirements. However, because it was entered into your corrective action program, the NRC is treating the finding as a noncited violation, consistent with Section 2.3.2 of the NRC Enforcement Policy. The NCV is described in the subject inspection report. If you contest the noncited violation or the significance of the noncited violation, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington DC 20555-0001, with copies to: (1) the Regional Administrator, Region IV; (2) the Director, Office of Enforcement, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and (3) the NRC Resident Inspector at

Cooper Nuclear Station. In addition, if you disagree with the characterization of any finding in this report, you should provide a response within 30 days of the date of this inspection report, with the basis for your disagreement, to the Regional Administrator, Region IV, and the NRC Resident Inspector at Cooper Nuclear Station. The information you provide will be considered in accordance with Inspection Manual Chapter 0305.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure(s), and your response, if you choose to provide one, will be made available electronically for public inspection in the NRC Public Document Room or from the NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html. To the extent possible, your response should not include any personal privacy or proprietary, information so that it can be made available to the Public without redaction.

Sincerely,

A. Vegel

Anton Vegel, Director
Division of Reactor Safety

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

Enclosure: Inspection Report No. 05000298/2010006
w/Attachments: Supplemental Information
Final Significance Determination Summary

cc w/enclosure:
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U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket:	50-298
License:	DPR-46
Report Nos.:	05000298/2010006
Licensee:	Nebraska Public Power District
Facility:	Cooper Nuclear Station
Location:	72676 648A Avenue Brownville, NE 68321
Dates:	October 18, 2010 through March 14, 2011
Team Leader:	J. Mateychick, Senior Reactor Inspector, Engineering Branch 2
Inspectors:	S. Alferink, Reactor Inspector, Engineering Branch 2 E. Uribe, Reactor Inspector, Engineering Branch 2 J. Watkins, Reactor Inspector, Engineering Branch 2 G. George, Reactor Inspector, Engineering Branch 1
Approved By:	Anton Vegel, Director Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000298/2010006; October 18, 2010 – March 14, 2011, Nebraska Public Power District; Cooper Nuclear Station: Triennial Fire Protection Team Inspection.

This report covers a two week fire protection team inspection, follow-up inspection and significance determination effort by specialist inspectors from Region IV. One finding was identified with an associated apparent violation, which was preliminary determined to have low-to-moderate safety significance (White). Two Green findings, which were noncited violations (NCVs), were also identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red) using Inspection Manual Chapter 0609, "Significance Determination Process." Findings for which the significance determination process (SDP) does not apply may be Green or be assigned a severity level after NRC management review. The crosscutting aspects, where applicable, were determined using Inspection Manual Chapter 0310, "Components Within the Cross Cutting Areas." The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 4, dated December 2006.

A. NRC-Identified and Self-Revealing Findings

Cornerstone: Mitigating Systems

- Apparent Violation. An apparent violation of 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and Criterion XVI, "Corrective Action," with a preliminary white significance, was identified for failure to ensure that some steps contained in Emergency Procedures at Cooper Nuclear Station would work as written and the concurrent failure to assure that a condition adverse to quality was promptly identified and corrected, respectively. Specifically, steps in Emergency Procedure 5.4 POST-FIRE, "Post-Fire Operational Information," and Emergency Procedure 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside Control Room," intended to reposition motor operated valves from the motor starter cabinet, would not have worked as written because the steps were not appropriate for the configuration of three valve motor starters. This finding was entered into the licensee's corrective action program under Condition Reports CR-CNS-2010-08193 and CR-CNS-2010-08242, however the licensee failed to adequately correct the procedure and the procedure remained unworkable.

  The failure to verify that procedure steps needed to safely shutdown the plant in the event of a fire would actually reposition motor operated valves to the required positions and the simultaneous failure to address the previous finding that the same procedure steps would not work as written, was a performance deficiency. This finding was more than minor safety significance because it impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. This finding affected both the procedure quality and protection against external factors (such as fires) attributes of this cornerstone objective. This finding was determined to have a preliminary low-to-moderate safety significance (White) during a Phase 3 evaluation using best available information. This problem, which has existed since 1997, involves risk factors that were not dependent on specific fire damage. The scenarios of concern involve larger fires in specific areas of the plant which trigger operators to implement fire response procedures to place the plant in a safe shutdown condition. Since some of those actions could not be completed using the procedures as written, this would challenge the operators' ability to establish adequate core cooling. This finding had a crosscutting aspect in the Corrective Action Program component, under the Problem Identification and Resolution area (P.1(c) - Evaluation), because the licensee failed to properly evaluate the circuit operation or conduct verification tests to ensure that corrective actions for a previous violation would reliably position the three valves. Upon identification of this issue, both emergency procedures were revised to assure correct valve alignment by manually operating the valve locally. Therefore, this finding does not represent a current safety concern. (Section 1R05.1)

- Green. A noncited violation of 10 CFR 50.65(a)(2) was identified for the failure to monitor the performance of the emergency lighting system against the established performance criteria. The licensee included the emergency lighting system in the Maintenance Rule program and specified that the emergency light batteries must be capable of 8 hours of operation, as required by 10 CFR Part 50, Appendix R, Section III.J. The team identified that the licensee did not perform tests that demonstrated the capability of the emergency lights to last for 8 hours; therefore, the licensee failed to monitor the performance of the emergency lights against the established performance criteria. This finding was entered into the licensee's corrective action program under Condition Reports CR-CNS-2010-08014 and CR-CNS-2010-08250.

  The failure to monitor the performance of the emergency lighting system against the performance criteria stated in the Maintenance Rule program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure to ensure that emergency lights would last for 8 hours could adversely affect the ability of operators to perform all of the manual actions required to support safe shutdown in the event of a fire. The significance of this finding was evaluated using Inspection Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," because the performance deficiency affected fire protection defense-in-depth strategies involving post fire safe shutdown systems. The finding was assigned a low degradation rating since the finding minimally impacted the performance and reliability of the fire protection program element. Specifically, the team determined that the licensee's preventive maintenance strategy provided reasonable assurance that the emergency lights would last sufficiently long for the operators to perform the most time-critical manual actions required to support safe shutdown in the event of a fire. The team also noted that operators were required to obtain and carry flashlights. Therefore, the finding screened as having very low safety significance (Green). This finding had a crosscutting aspect in the area of Human Performance associated with Decision Making because the licensee failed to identify possible unintended consequences of the decision to change the maintenance program for the emergency lights. Specifically, the licensee failed to identify that deleting emergency light testing impacted Maintenance Rule performance monitoring. [H.1(b)] (Section 1R05.8)

B. Licensee-Identified Violations

None

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R05 Fire Protection (71111.05TTP)

This report presents the results of a triennial fire protection inspection conducted in accordance with NRC Inspection Procedure 71111.05TTP, "Fire Protection-NFPA Transition Period (Triennial)," at Cooper Nuclear Station. The licensee committed to adopt a risk informed fire protection program in accordance with National Fire Protection Association Standard 805 (NFPA-805), but had not yet completed the program transition. The inspection team evaluated the implementation of the approved fire protection program in selected risk-significant areas, with an emphasis on the procedures, equipment, fire barriers, and systems that ensure the post-fire capability to safely shut the plant down.

Inspection Procedure 71111.05TTP requires selecting three to five fire areas for review. The inspection team used the fire hazards analysis section of the Cooper Nuclear Station Individual Plant Examination of External Events to select the following five risk-significant fire zones (inspection samples) for review:

- Fire Area I / Fire Zone 2A — Control Rod Drive Units - North
Reactor Building Elevation 903' 6"
- Fire Area I / Fire Zone 5B — Reactor Motor Generator Set Area
Reactor Building Elevation 976' 0"
- Fire Area II / Fire Zone 3A — Switchgear Room 1F
Reactor Building Elevation 931' 6"
- Fire Area IX / Fire Zones 14A — Diesel Generator 1A Room
Diesel Generator Building Elevation 903' 6"
- Fire Area IX / Fire Zones 14C — Diesel Oil Day Tank Room
Diesel Generator Building Elevation 903' 6"

The inspection team evaluated the licensee's fire protection program using the applicable requirements, which included plant Technical Specifications, Operating License Condition 2.C.(5); NRC safety evaluations; 10 CFR 50.48; Branch Technical Position 9.5-1; and 10 CFR 50, Appendix R. The team also reviewed related documents that included the Final Safety Analysis Report (FSAR), Section 9.5; the fire hazards analysis; and the post-fire safe shutdown analysis.

Specific documents reviewed by the team are listed in the attachment. Five fire area inspection samples were completed. Also, one B.5.b strategy review sample was completed.

.1 Protection of Safe Shutdown Capabilities

a. Inspection Scope

The team reviewed the piping and instrumentation diagrams, safe shutdown equipment list, safe shutdown design basis documents, and the post fire safe shutdown analysis to verify that the licensee properly identified the components and systems necessary to achieve and maintain safe shutdown conditions for fires in the selected fire areas. The team observed walkdowns of the procedures used for achieving and maintaining safe shutdown in the event of a fire to verify that the procedures properly implemented the safe shutdown analysis provisions.

For each of the selected fire areas, the team reviewed the separation of redundant safe shutdown cables, equipment, and components located within the same fire area. The team also reviewed the licensee's method for meeting the requirements of 10 CFR 50.48; Branch Technical Position 9.5-1, Appendix A; and 10 CFR Part 50, Appendix R, Section III.G. Specifically, the team evaluated whether at least one post-fire safe shutdown success path would remain free of fire damage in the event of a fire. In addition, the team verified that the licensee met applicable license commitments.

b. Findings

Introduction. An apparent violation of 10 CFR Part 50, Appendix B, Criterion V and Criterion XVI, with a preliminary White significance, was identified for the repeated failure to ensure that some steps contained in emergency procedures at Cooper Nuclear Station would work as written. Specifically, steps in Emergency Procedure 5.4 POST-FIRE, "Post Fire Operational Information," and Emergency Procedure 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside Control Room," intended to reposition motor operated valves at the motor starter cabinet, would not have worked as written because the steps were not appropriate for the configuration of the motor starters.

Description. Post-fire safe shutdown strategies at the Cooper Nuclear Station require equipment operations to be performed in accordance with one of two emergency procedures. For most fire areas, plant shutdown is performed using Emergency Procedure 5.4 POST-FIRE, "Post-Fire Operational Information," Revision 37, in conjunction with other plant procedures. For areas where fires might necessitate evacuation of the control room, alternative shutdown is performed using Emergency Procedure 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside the Control Room," Revision 38.

The team performed a walkthrough of Emergency Procedure 5.4 POST-FIRE for selected fire areas by observing plant operators simulate actions required by the procedure. This procedure required operators to reposition multiple motor-operated valves (MOVs) from each valve's motor starter cabinet. The procedure steps direct operators to open the motor starter cabinet, remove the control power fuses, then press designated contactors for a specified amount of time to reposition the valve to the required position.

The team was concerned that some of the procedure steps might not be reliably performed by the operators because bulky electrical safety gloves might not allow access to recessed contactors. When the licensee attempted to demonstrate their method, they identified that it would not work for one type of contactor. The internal configuration of the contactor would not complete the power circuit by depressing it. The manufacturer describes the design as having "direct magnet drive with positive pull-in of contactors." Since control power was removed by pulling fuses before operating the contactors, the magnet system would not engage the power contacts to the valve motor. The inspectors noted that the operator performing the procedure steps would have no indication that the valve(s) did not reposition. Because the procedures do not specifically require checking the valve positions for most fire locations, the failure to reposition would not be readily apparent.

The three valves with this type of contactor were residual heat removal (RHR) system valves RHR-MO-25A and RHR-MO-25B, Train A and B Inboard Injection Isolation Valves, and reactor recirculation (RR) system valve RR-MO-53A, Reactor Recirculation A Pump Discharge Valve. The procedural deficiency in Emergency Procedure 5.4 POST-FIRE impacted the response to fires in 11 fire areas, each involving one valve. One of the valves, RHR-MO-25B, is operated in the same manner during alternative shutdown in accordance with Emergency Procedure 5.4 FIRE-S/D, which contained the same procedural deficiency, for fires in two additional fire areas. The 13 affected fire areas are listed below:

<u>Fire Area</u>

CB-A	Control Building Reactor Protection System Room 1A, Seal Water Pump Area, and Hallway
CB-A-1	Control Building Division 1 Switchgear Room and Battery Room
CB-B	Control Building Division 2 Switchgear Room and Battery Room
CB-C	Control Building Reactor Protection System Room 1B
CB-D	Control Room, Cable Spreading Room, Cable Expansion Room, and Auxiliary Relay Room
RB-DI (SE)	Reactor Building RHR Pump B/HPCI Pump Room
RB-DI (SW)	Reactor Building South/Southwest 903, Southwest Quad 889 and 859, and RHR Heat Exchanger Room B
RB-FN	Reactor Building 903, Northeast Corner
RB-J	Reactor Building Critical Switchgear Room 1F
RB-K	Reactor Building Critical Switchgear Room 1G
RB-M	Reactor Building North/Northwest 931 and RHR Heat Exchanger Room A
RB-N	Reactor Building South/Southwest 931 and RHR Heat Exchanger Room B
TB-A	Turbine Building (multiple areas)

Opening either valve RHR-MO-25A or valve RHR-MO-25B is necessary to establish alternative shutdown cooling. Alternative shutdown cooling involves using a train of RHR to take suction from the suppression pool, inject the low pressure water to flood the reactor vessel, and recirculate the water through the safety relief valves (SRVs) back to the suppression pool. Establishing alternative shutdown cooling can be very time-sensitive. If high-pressure coolant injection (HPCI) is not available, the licensee provided calculations that show that core damage can occur in as little as 15 minutes after valve RHR-MO-25B fails to open.

Valve RR-MO-53A is the discharge isolation valve for Reactor Recirculation Pump 1-A. This valve is only required for cold shutdown. For some fire areas, the normal shutdown cooling mode of RHR system operation was credited in the fire safe shutdown analysis to be available. In shutdown cooling mode, the RHR system takes suction from the suction pipe of reactor recirculation system loop "A". The reactor coolant is then cooled and returned to a reactor recirculation loop discharge pipe. The failure to close either valve RR-MO-53A or RR-MO-43A would result in a short circuit of the shutdown cooling flow, bypassing the reactor vessel. The cool down from hot shutdown conditions and the transition to normal shutdown cooling allows time to close either valve RR-MO-53A or RR-MO-43A using local manual operation.

In 2004, a related but separate violation (NCV 05000298/2004008-01) was issued for failure to protect cables from fire damage for MOVs required to be available for post fire safe shutdown. The licensee committed to adopt a risk-informed fire protection program in accordance with 10 CFR 50.48(c) and NFPA-805, and planned to address the 2004 violation through their NFPA-805 conversion. To be able to delay correcting the 2004 violation, the licensee was required to verify that the compensatory measures for the violation (the operator manual actions) were adequate to ensure safety, in this case to be able to safely shut the plant down in the event of a fire.

Inspection Report 05000298/2004008 noted reliability concerns with the method of operating the MOVs. These included the fact that the contactors were not labeled to allow operators to know which contactors the procedure instructed them to operate, no indication was available at the motor starter cabinet for the operator to know the valves had reached their required position, and valve position was not verified locally at the valves. As part of corrective action, the licensee installed "open" and "closed" labels near contactors in the motor starter cabinets.

In 2007, inspectors identified that some of the operator manual actions used as compensatory measures for the 2004 violation would not have repositioned 10 of the MOVs. The procedures did not account for the fact that these 10 MOVs had different motor starter circuits than most valves. Despite installing labels following the 2004 violation, the licensee failed to recognize that these 10 MOVs had a more complex circuit design which required two or three contactors to be operated at the same time, while the procedures only required operating one "open" or one "close" contactor. A White finding with an associated violation (Violation 05000298/2008008-01, EA 07-204) was issued for having an inadequate procedure and failing to verify that the procedure would work.

Inspection Report 05000298/2008007 again documented the reliability concerns that there were no valve position indications at the MOV motor starter cabinets, and the procedures did not direct local valve position checks. Additional reliability concerns were also documented concerning the adequacy of the procedures and the instrumentation available to diagnose the failure of an MOV to reposition.

The licensee took corrective actions to change and verify the procedures to address the 2008 finding; however the licensee's efforts again failed to identify details of the electrical design which would result in the procedure steps not repositioning three MOVs.

Analysis. The failure to verify that procedure steps needed to safely shutdown the plant in the event of a fire would actually reposition motor operated valves to the required positions, and to address a previous finding that the same procedure steps would not work as written, was a performance deficiency. This performance deficiency is of more than minor safety significance because it impacted the Mitigating Systems cornerstone objective to ensure the availability, reliability, and capability of systems that respond to external events (such as fire) to prevent undesirable consequences. This finding affected both the procedure quality and protection against external factors (such as fires) attributes of this cornerstone objective.

The significance determination process (SDP) Phase 1 Screening Worksheet (Manual Chapter 0609, Attachment 4), Table 3b directs the user to Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," because it affected fire protection defense-in-depth strategies involving post fire safe shutdown systems. However, the Assumptions and Limitations section of Appendix F states that finings involving multiple fire areas are beyond the scope of Appendix F, and findings involving control room evacuation are not explicitly treated in Appendix F. Therefore, a Phase 3 analysis was performed.

The license claimed that the issue involved a performance deficiency that only impacted cold shutdown, and therefore should be screened as Green during a Phase 1 SDP. The NRC concluded that this finding cannot be screened out because the complexity of the issue (e.g., multiple fire areas affected) precludes simple screening, and because the plant conditions and system dependencies prevent a conclusion that only cold shutdown is affected.

Manual Chapter 0308 describes the basis for Appendix F screening out issues involving only cold shutdown as follows:

> The second question screens findings to green that impact only the ability of the plant to achieve cold shutdown. This is consistent with the common risk analysis practice of defining hot shutdown as success. That is, both fire PRAs [probabilistic risk assessments] and Internal Events PRAs typically assume that achieving a safe and stable hot shutdown state constitutes success and the end state for accident sequence analyses. Note that this screening step applies only to findings against 10CFR50 Appendix R, Section III.G.1.b. All other regulatory provisions are considered to involve, in part or in whole, measures provided for preservation and protection of the post-fire hot shutdown capability and will not be screened in this step (e.g., fire prevention, fire suppression, fire brigade, fire barriers, etc.).

The licensee's fire safe shutdown strategy and implementing procedures for the scenarios of concern direct operators to proceed to cold shutdown within a few hours. Operation in hot shutdown and cold shutdown rely on the suppression pool with limited capability for cooling the suppression pool. This strategy is too complex to allow simple risk screening for this finding.

A risk analysis was performed previously for the 2008 procedural problems that affected ten valves, including the three valves addressed by this performance deficiency. This was documented in Inspection Report 05000298/2008008 (EA 07-204). In both the 2008 and current cases, valves RHR-MOV-25A, RHR-MOV-25B, and RHR-MOV-53A were incapable of being remotely operated from the motor starter as prescribed by Procedures 5.4 POST-FIRE and 5.4 FIRE-S/D. Therefore, the linked event tree model developed for the risk estimate performed in 2008 was used to assess the significance of the current issue for these three valves.

Fires that do not require control room evacuation are addressed in Procedure 5.4 POST-FIRE. For fire areas that do not involve control room evacuation, the analyst concluded that the risk for the current finding is less than 1.0E-7 (this is unchanged from 2008 evaluation).

The risk attributable to post fire remote shutdown (control room abandonment sequences) results predominantly from the failure of Valve RHR-MOV-25B to open as described in Procedure 5.4 FIRE-S/D. This is the credited train and the only procedural means for initiating alternative shutdown cooling during the recovery actions. Changes were made to Procedure 5.4 FIRE-S/D subsequent to the 2008 issue which were credited in the current analysis and resulted in a decrease in the risk significance of the subject valves.

The non-recovery probability was decreased by a factor of 78 for the current finding because of changes that were made to Procedure 5.4 FIRE-S/D. These changes in Attachment 1 of the procedure directed the operator at the remote shutdown panel to close SRVs if RHR injection was not observed to be successful and stabilize conditions using high pressure injection. Also, it directed operators to delay securing HPCI (if it was running) until RHR injection is confirmed. Additionally, Attachment 2 to the procedure directed the reactor building operator to open valve RHR-MOV-25B manually if the valve did not operate. However, there is limited instrumentation available at the remote shutdown panel to be able to recognize and diagnose that the valve did not open, and no available indications at the motor starter cabinet. Therefore, the operator who might be able to diagnose the failure of RHR-MO-25B did not have a procedure with the critical recovery step, and the operator with the correct recovery step in his procedure did not have the capability to know whether it was needed.

Using the linked event tree model and a period of exposure of one year, the analyst calculated the ΔCDF to be 2.0E-6/yr for postulated fires leading to the abandonment of the main control room. The analyst concluded that the performance deficiency was of low to moderate significance (White).

A more detailed description to the Phase 3 analysis is attached to this report.

The NRC expects that licensees will ensure that issues potentially impacting nuclear safety are promptly identified, fully evaluated, and that actions are taken to address safety issues in a timely manner, commensurate with their significance. Additionally, the NRC expects that for significant problems, licensees will conduct effectiveness reviews of corrective actions to ensure that the problems are resolved. Because the licensee failed to properly evaluate the circuit operation or conduct verification tests to ensure that corrective actions for a previous violation would reliably position the three valves, the team concluded that this finding has a crosscutting aspect in the Corrective Action Program component, under the Problem Identification and Resolution area (P.1(c) - Evaluation).

Enforcement. Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," requires, in part, that activities affecting quality shall be prescribed by documented instructions, procedures, or drawings, of a type appropriate to the circumstances and shall be accomplished in accordance with these instructions, procedures, or drawings.

Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI requires, in part:

> 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.

Emergency Procedure 5.4 POST-FIRE, "Post-Fire Operational Information," Revision 37, and Emergency Procedure 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside the Control Room," Revision 38, were designated as quality-related procedures used to implement operator actions to safely shutdown the plant in response to a fire. Violation 05000298/2008008-01 (EA 07-204) documented a significant condition adverse to quality in that steps in Emergency Procedure 5.4 POST-FIRE and Emergency Procedure 5.4 FIRE-S/D would not achieve and maintain a safe shutdown condition in the event of certain fires.

Contrary to the above, between July 1997 and November, 2010, the licensee failed to ensure that activities affecting quality were prescribed by documented procedures appropriate to the circumstances, and to assure that a significant condition adverse to quality was promptly corrected. Specifically, Emergency Procedure 5.4 POST-FIRE and Emergency Procedure 5.4 FIRE-S/D were changed in 1997 to add steps that were inappropriate to the circumstances because they would not work as written to reposition three motor operated valves needed to establish core cooling. The licensee failed to properly verify and validate procedure steps when the procedure changes were made and on multiple occasions between July 1997 and November 2010, including verification and validation actions performed in response to Violation 05000298/2008008-01..

In addition, contrary to the above, between July 2008 and November 2010, the licensee failed to identify, correct, and preclude repetition of a significant condition adverse to quality. Specifically, Violation 05000298/2008008-01 identified a significant condition adverse to quality in that Emergency Procedure 5.4 POST-FIRE and Emergency Procedure 5.4 FIRE-S/D would not work as written and the licensee had failed to verify and validate procedure steps to ensure that they would work to accomplish the necessary tasks. While addressing that violation, the licensee failed to perform sufficient circuits to identify and correct a problem with valves RHR-MOV-25A, RHR-MOV-25B, and RHR-MOV-53A.

The licensee entered this issue into their corrective action program as Condition Reports CR-CNS-2010-08193 and CR-CNS-2010-08242. This violation is being treated as an apparent violation (AV), consistent with the Enforcement Policy: AV 05000298/2010006-01, Inadequate Post-Fire Safe Shutdown Procedures.

Because the licensee failed to correct this condition as part of Violation 05000298/2008008-01, and because Violation 05000298/2008008-01 did not receive enforcement discretion, this finding was not appropriate for enforcement discretion.

.2 Passive Fire Protection

a. Inspection Scope

The team walked down accessible portions of the selected fire areas to observe the material condition and configuration of the installed fire area boundaries (including walls, fire doors, and fire dampers) and verify that the electrical raceway fire barriers were appropriate for the fire hazards in the area. The team compared the installed configurations to the approved construction details, supporting fire tests, and applicable license commitments.

The team reviewed installation, repair, and qualification records for a sample of penetration seals to ensure that the fill material possessed an appropriate fire rating and that the installation met the engineering design. The team also reviewed similar records for the rated fire wraps to ensure the material possessed an appropriate fire rating and that the installation met the engineering design.

b. Findings

No findings were identified.

.3 Active Fire Protection

a. Inspection Scope

The team reviewed the design, maintenance, testing, and operation of the fire detection and suppression systems in the selected fire areas. The team verified that the manual and automatic detection and suppression systems were installed, tested, and maintained in accordance with the National Fire Protection Association code of record or approved deviations, and that each suppression system was appropriate for the hazards in the selected fire areas.

The team performed a walkdown of accessible portions of the detection and suppression systems in the selected fire areas. The team also performed a walkdown of major system support equipment in other areas (e.g., fire pumps) to assess the material condition of these systems and components.

The team reviewed the electric and diesel fire pump flow and pressure tests to verify that the pumps met their design requirements. The team also reviewed high pressure carbon dioxide suppression system functional tests and inspections to verify that the system capability met the design requirements.

The team assessed the fire brigade capabilities by reviewing training, qualification, and drill critique records. The team also reviewed pre-fire plans and smoke removal plans for the selected fire areas to determine if appropriate information was provided to fire brigade members and plant operators to identify safe shutdown equipment and instrumentation, and to facilitate suppression of a fire that could impact post-fire safe shutdown capability. In addition, the team inspected fire brigade equipment to determine operational readiness for fire fighting.

The team observed an unannounced fire drill, conducted on November 1, 2010, and the subsequent drill critique using the guidance contained in Inspection Procedure 71111.05AQ, "Fire Protection Annual/Quarterly." The team observed fire brigade members fight a simulated fire in the Reactor Building, located in a switchgear room. The team verified that the licensee identified problems, openly discussed them in a self-critical manner at the drill debrief, and identified appropriate corrective actions. Specific attributes evaluated were: (1) proper wearing of turnout gear and self-contained breathing apparatus; (2) proper use and layout of fire hoses; (3) employment of appropriate fire fighting techniques; (4) sufficient fire fighting equipment was brought to the scene; (5) effectiveness of fire brigade leader communications, command, and control; (6) search for victims and propagation of the fire into other areas; (7) smoke removal operations; (8) utilization of pre-planned strategies; (9) adherence to the pre-planned drill scenario; and (10) drill objectives.

b. Findings

No findings were identified.

.4 Protection From Damage From Fire Suppression Activities

a. Inspection Scope

The team performed plant walkdowns and document reviews to verify that redundant trains of systems required for hot shutdown, which are located in the same fire area, would not be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems. Specifically, the team verified that:

- A fire in one of the selected fire areas would not directly, through production of smoke, heat, or hot gases, cause activation of suppression systems that could potentially damage all redundant safe shutdown trains.
- A fire in one of the selected fire areas or the inadvertent actuation or rupture of a fire suppression system would not directly cause damage to all redundant trains.
- Adequate drainage was provided in areas protected by water suppression systems.

b. Findings

No findings were identified.

.5 Alternative Shutdown Capability

a. Inspection Scope

Review of Methodology

The team reviewed the safe shutdown analysis, operating procedures, piping and instrumentation drawings, electrical drawings, the Final Safety Analysis Report, and other supporting documents to verify that hot and cold shutdown could be achieved and maintained from outside the control room for fires that require evacuation of the control room, with or without offsite power available.

Plant walkdowns were conducted to verify that the plant configuration was consistent with the description contained in the safe shutdown and fire hazards analyses. The team focused on ensuring the adequacy of systems selected for reactivity control, reactor coolant makeup, reactor decay heat removal, process monitoring instrumentation, and support systems functions.

The team also verified that the systems and components credited for shutdown would remain free from fire damage. Finally, the team verified that the transfer of control from the control room to the alternative shutdown location would not be affected by fire-induced circuit faults (e.g., by the provision of separate fuses and power supplies for alternative shutdown control circuits).

Review of Operational Implementation

The team verified that licensed and non-licensed operators received training on alternative shutdown procedures. The team also verified that sufficient personnel to perform a safe shutdown were trained and available onsite at all times, exclusive of those assigned as fire brigade members.

A walkthrough of the post fire safe shutdown procedure with licensed and non-licensed operators was performed to determine the adequacy of the procedure. The team verified that the operators could be reasonably expected to perform specific actions within the time required to maintain plant parameters within specified limits. Time critical actions that were verified included restoring electrical power, establishing control at the remote shutdown and local shutdown panels, establishing reactor coolant makeup, and establishing decay heat removal.

The team reviewed manual actions to ensure that they had been properly reviewed and approved and that the actions could be implemented in accordance with plant procedures in the time necessary to support the safe shutdown method for each fire area.

The team also reviewed the periodic testing of the alternative shutdown transfer capability and instrumentation and control functions to verify that the tests are adequate to demonstrate the functionality of the alternative shutdown capability.

b. Findings

No findings were identified.

.6 Circuit Analysis

a. Inspection Scope

This segment of inspection is suspended for plants in transition to a risk-informed fire protection program in accordance with NFPA 805. Therefore, the team did not evaluate this area.

b. Findings

No findings were identified.

.7 Communications

a. Inspection Scope

The team inspected the contents of designated emergency storage lockers and reviewed the alternative shutdown procedure to verify that portable radio communications and fixed emergency communications systems were available, operable, and adequate for the performance of designated activities. The team verified the capability of the communication systems to support the operators in the conduct and coordination of their required actions. The team also verified that the design and location of communications equipment such as repeaters and transmitters would not cause a loss of communications during a fire. The team discussed system design, testing, and maintenance with the system engineer.

The team reviewed the licensee's response to Condition Report CR-CNS-2010-07848. The team verified the licensee properly implemented the Maintenance Rule program with respect to the communications systems required for alternative shutdown.

b. Findings

No findings were identified.

.8 Emergency Lighting

a. Inspection Scope

The team reviewed the portion of the emergency lighting system required for alternative shutdown to verify that it was adequate to support the performance of manual actions required to achieve and maintain hot shutdown conditions and to illuminate access and egress routes to the areas where manual actions would be required. The team evaluated the locations and positioning of the emergency lights during a walkthrough of the alternative shutdown procedure.

The team verified that the licensee installed emergency lights with an 8-hour capacity, maintained the emergency light batteries in accordance with manufacturer recommendations, and tested and performed maintenance in accordance with plant procedures and industry practices. The team also verified the licensee properly implemented the Maintenance Rule program with respect to the emergency lighting systems required for alternative shutdown.

The team identified several concerns with the adequacy of the emergency lights during the walkthrough of the alternative shutdown procedure. In response to these concerns, the licensee performed blackout tests to demonstrate the adequacy of the installed emergency lights. The team observed blackout tests in the following areas:

- Control Building Corridor, 903' Elevation
- Control Building Basement, 881' Elevation
- Diesel Generator 2 Room

b. Findings

Introduction. The team identified a Green noncited violation of 10 CFR 50.65(a)(2) for the failure to monitor the performance of the emergency lighting system against the established performance criteria.

Description. During the inspection, the team reviewed the licensee's maintenance program for the emergency lighting system. The team determined that the licensee did not perform tests that demonstrated the capability of the emergency lights to last 8 hours. Instead, the licensee replaced each emergency light battery at a prescribed frequency. The licensee previously demonstrated the capability of the emergency lights to last 8 hours via the performance of internal resistance measurements. In 2008, the licensee modified their maintenance program to remove the internal resistance measurements and rely upon the prescribed replacement strategy.

The team also reviewed the licensee's implementation of their Maintenance Rule program with respect to the emergency lighting system. The licensee included the emergency lighting system into the Maintenance Rule program and included a performance criterion for the emergency light batteries to support 8-hours of operation, as required by 10 CFR Part 50, Appendix R, Section III.J.

Since the licensee did not perform tests that demonstrated the capability of the emergency lights to last 8 hours, the team determined that the licensee failed to monitor the performance of the emergency lights against the established performance criteria.

Analysis. The failure to monitor the performance of the emergency lighting system against the performance criteria stated in the Maintenance Rule program was a performance deficiency. The performance deficiency was more than minor because it was associated with the protection against external events (fire) attribute of the Mitigating Systems Cornerstone and it adversely affected the cornerstone objective of ensuring the availability, reliability, and capability of systems that respond to initiating events to prevent undesirable consequences. Specifically, the failure of the emergency lights to last 8 hours could adversely affect the ability of operators to perform the manual actions required to support safe shutdown in the event of a fire.

The significance of this finding was evaluated using Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," because the performance deficiency affected fire protection defense-in-depth strategies involving post-fire safe shutdown systems. The team assigned the performance deficiency to the Post-fire Safe Shutdown category since it affected systems or functions relied upon for post-fire safe shutdown.

The finding was assigned a low degradation rating since the finding minimally impacted the performance and reliability of the fire protection program element. Specifically, the team determined that the licensee's preventive maintenance strategy provided reasonable assurance that the emergency lights would last sufficiently long for the operators to perform the most time critical manual actions required to support safe shutdown in the event of a fire. The team also noted that operators were required to obtain and carry flashlights. Therefore, the finding screened as having very low safety significance (Green).

The NRC expects that licensee decisions demonstrate that nuclear safety is an overriding priority and to conduct effectiveness reviews of safety-significant decisions to identify possible unintended consequences. Because the licensee failed to identify that deleting emergency light testing impacted Maintenance Rule performance monitoring, the team concluded that this finding had a crosscutting aspect in the area of human performance associated with decision making. Specifically, the licensee failed to identify possible unintended consequences of the decision to change the maintenance program for the emergency lights. [H.1(b)]

<u>Enforcement</u>. Title 10 of the Code of Federal Regulations, Part 50, Section 65, Paragraph (a)(1), requires, in part, that licensees shall monitor the performance or conditions of structures, systems, or components (SSCs) within the scope of the maintenance rule as defined by 10 CFR 50.65 (b), against licensee established goals, in a manner sufficient to provide reasonable assurance that such SSCs are capable of fulfilling their intended functions.

Title 10 of the Code of Federal Regulations, Part 50, Section 65, Paragraph (a)(2) states, in part, that monitoring as specified in 10 CFR 50.65 (a)(1) is not required where it has been demonstrated that the performance or condition of a SSC is being effectively controlled through the performance of appropriate preventive maintenance, such that the SSC remains capable of performing its intended function.

The licensee's Maintenance Rule program included the emergency lighting system and established a performance criterion that the emergency lighting system batteries support 8-hours of operation, as required by 10 CFR Part 50, Appendix R, Section III.J.

Contrary to the above, from October 3, 2008 to November 5, 2010, the licensee failed to demonstrate that the performance of the emergency lighting system was effectively controlled through the performance of appropriate preventive maintenance and did not smonitor the emergency lighting system against licensee established goals. Specifically, the licensee failed to demonstrate that the emergency lighting system remained capable of providing 8 hours of illumination for post-fire safe shutdown.

The licensee entered this issue into their corrective action program as Condition Reports CR-CNS-2010-08014 and CR-CNS-2010-08250. Because this violation was of very low safety significance and it was entered into the licensee's corrective action program, this violation is being treated as a noncited violation, consistent with the Enforcement Policy: NCV 05000298/2010006-03, Failure to Monitor the Performance of the Emergency Lights Against the Maintenance Rule Criteria.

.9 Cold Shutdown Repairs

a. Inspection Scope

The team verified that the licensee identified repairs needed to reach and maintain cold shutdown and had dedicated repair procedures, equipment, and materials to accomplish these repairs. Using these procedures, the team evaluated whether these components could be repaired in time to bring the plant to cold shutdown within the time frames specified in the design and licensing bases. The team verified that the repair equipment, components, tools, and materials needed for the repairs were available and accessible on site.

b. Findings

No findings were identified.

.10 Compensatory Measures

a. Inspection Scope

The team verified that compensatory measures were implemented for out-of-service, degraded, or inoperable fire protection and postfire safe shutdown equipment, systems, or features (e.g., detection and suppression systems and equipment; passive fire barriers; or pumps, valves, or electrical devices providing safe shutdown functions). The team also verified that the short-term compensatory measures compensated for the degraded function or feature until appropriate corrective action could be taken and that the licensee was effective in returning the equipment to service in a reasonable period of time.

b. Findings

A finding related to this review was documented in Section 1R05.01. No additional findings were identified.

.11 B.5.b Inspection Activities

a. Inspection Scope

The team reviewed the licensee's implementation of guidance and strategies intended to maintain or restore core, containment, and spent fuel pool cooling capabilities under the circumstances associated with loss of large areas of the plant due to explosions or fire as required by Section B.5.b of the Interim Compensatory Measures Order, EA-02-026, dated February 25, 2002 and 10 CFR 50.54(hh)(2).

The team reviewed a licensee's strategy to verify that they continued to maintain and implement procedures, maintain and test equipment necessary to properly implement the strategy, and to ensure that station personnel are knowledgeable and capable of implementing the procedure. The team performed a visual inspection of portable equipment used to implement the strategy to ensure availability and material readiness of the equipment, including the adequacy of portable pump trailer hitch attachments, and verify the availability of onsite vehicles capable of towing the portable pump. The team assessed the offsite ability to obtain fuel for the portable pump, and foam used for firefighting efforts. The team reviewed the following strategy as an inspection sample:

- 5.3 Alt-Strategy, "Alternative Core Cooling Mitigating Strategies," Revision 023, Attachment 4, "Manual Operation of RCIC [reactor core isolation cooling]."

b. Findings

No findings were identified.

4. OTHER ACTIVITIES [OA]

4OA2 Identification and Resolution of Problems

Corrective Actions for Fire Protection Deficiencies

a. Inspection Scope

The team selected a sample of condition reports associated with the licensee's fire protection program to verify that the licensee had an appropriate threshold for identifying deficiencies. In addition, the team reviewed the corrective actions proposed and implemented to verify that they were effective in correcting identified deficiencies. The team also evaluated the quality of recent engineering evaluations through a review of condition reports, calculations, and other documents during the inspection.

b. Findings

Findings related to this review are documented in Sections 1R05.01 and 1R05.05. No additional findings were identified.

4OA6 Meetings, Including Exit

Exit Meeting Summary

The team presented the inspection results to Mr. D. Willis, General Manager, Plant Operations, and other members of the licensee staff at a debrief meeting on November 5, 2010. The licensee acknowledged the findings presented.

The team presented the inspection results to Mr. D. Buman, Director of Engineering, and other members of the licensee staff at an exit meeting on March 14, 2011. The licensee acknowledged the findings presented.

The inspectors confirmed that proprietary material examined during the inspection had been returned.

ATTACHMENTS: SUPPLEMENTAL INFORMATION
FINAL SIGNIFICANCE DETERMINATION SUMMARY

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

J. Aldana, Security Coordinator
R. Alexander, Electrical Superintendent
J. Austin, System Engineering Manager
T. Barker, Quality Assurance Manager
J. Bebb, Security Manager
S. Bebb, Administrative Services Manager
M. Bergmeier, Operation Support Group Supervisor
K. Billesbach, Materials, Purchasing and Contracts Manager
D. Buman, Director of Engineering
K. Cardy, Fire Protection Engineer
G. Chinn, Contractor
L. Deuhirst, Corrective Actions and Assessments Manager
R. Dyer, Engineering Support Program Engineer
J. Dykstra, Electrical Engineering Program Supervisor
R. Estrada, Design Engineering Manager
J. Flaherty, Senior Staff Licensing Engineer
J. Gage, Reactor Operator
R. Gauchat, Security Training Supervisor
T. Hattovy, Engineering Support Manager
D. Jones, Safety Coordinator
T. Kahland, Reactor Operator
C. Long, Engineering Specialist
D. McGargill, Non-Licensed Operator
T. Mueller, Senior Reactor Operator
K. Newcomb, Fire Marshal
D. Oshlo, Information Technology Manager
R. Penfield, Operations Manager
D. Seylock, Training Manager
J. Shrader, Fire Safety Lead, Nebraska Public Power District
D. Van Der Kap, Licensing Manager
M. Van Winkle, Electrical Design Supervisor
D. Weniger, Valves Program Engineer
D. Willis, General Manager, Plant Operations
A. Zaremba, Director of Nuclear Safety Assessment

NRC personnel

M. Chambers, Resident Inspector
S. Vaughn, NRR/DIRS/IPAB
J. Bowen, NRR/DIRS/IRIB
D. Loveless, Senior Reactor Analyst, RIV/DRS
M. Runyan, Senior Reactor Analyst, RIV/DRS

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000298/2009006-01	AV	Inadequate Post-Fire Safe Shutdown Procedures (Section 1R05.01)

Opened and Closed

05000298/2009006-02	NCV	Failure to Correct a Condition Adverse to Quality Related to Post-Fire Safe Shutdown (Section 1R05.05)

Closed None

LIST OF ACRONYMS

ADAMS	Agencywide Documents Access and Management System
BWR	Boiling Water Reactor
CR	Condition Report
CFR	Code of Federal Regulations
DRS	Division of Reactor Safety
FSAR	Final Safety Analysis Report
HPCI	High Pressure Coolant Injection
LPSI	Low Pressure Safety Injection
MOV	Motor Operated Valve
NCV	Noncited Violation
NFPA	National Fire Protection Association
NRC	Nuclear Regulatory Commission
PAR	Publicly Available Records
PRA	Probabilistic Risk Assessment
RCIC	Reactor Core Isolation Cooling
RHR	Residual Heat Removal
SDP	Significance Determination Process
SRV	Safety/Relief Valve

LIST OF DOCUMENTS REVIEWED

CALCULATIONS

Number	Title	Revision
NEDC 01-030	HPCI Room Heatup During Appendix R Shutdown from Alternative Shutdown Panel	2
NEDC 09-080	Multiple Spurious Operation Expert Panel Results	0
NEDC 85-081	Pressure Drop in Steam Line to the HPCI Turbine	0C1
NEDC 94-034H	Containment Analysis for Appendix R – Shutdown from Alternative Shutdown Room	2
NEDC 95-003	Determination of Allowable Operating Parameters for CNS MOV Program MOVs	23

CONDITION REPORTS (CRs)

CR-CNS-2004-03595	CR-CNS-2004-05511	CR-CNS-2006-03138
CR-CNS-2007-01248	CR-CNS-2007-04155	CR-CNS-2007-07065
CR-CNS-2008-05653	CR-CNS-2008-5751	CR-CNS-2008-05766
CR-CNS-2007-08253	CR-CNS-2010-02387	CR-CNS-2010-03500
CR-CNS-2010-05023	CR-CNS-2010-05269	CR-CNS-2010-05855
CR-CNS-2010-05856	CR-CNS-2010-06942	CR-CNS-2010-06184
CR-CNS-2010-06236	CR-CNS-2010-06245	CR-CNS-2010-06258
CR-CNS-2010-06264	CR-CNS-2010-06441	CR-CNS-2010-06775
CR-CNS-2010-06942	CR-CNS-2010-07010	CR-CNS-2010-07527
CR-CNS-2010-07527	CR-CNS-2010-07553	CR-CNS-2010-07553
CR-CNS-2010-07757*	CR-CNS-2010-07762*	CR-CNS-2010-07776*
CR-CNS-2010-07803*	CR-CNS-2010-07813*	CR-CNS-2010-07823*
CR-CNS-2010-07831*	CR-CNS-2010-07839*	CR-CNS-2010-07847*
CR-CNS-2010-07848*	CR-CNS-2010-07857*	CR-CNS-2010-07859*
CR-CNS-2010-07861*	CR-CNS-2010-07914*	CR-CNS-2010-08163*
CR-CNS-2010-08165*	CR-CNS-2010-08166*	CR-CNS-2010-08167*
CR-CNS-2010-08201*	CR-CNS-2010-08221*	CR-CNS-2010-08250*

CR-CNS-2010-08253*		

* Condition Report initiated due to inspection activities.

DRAWINGS

Number	Title	Revision
14EK-0144	Diesel Engine Generator Schematic Diagram	N22
85B-70008 Sheet 159	Wiring Diagram WD-12, 13, & 14 F.V.R. Starter	N00
0709-003	Ruskin Model NIBD23 3 Hour Type C – U.L. Labeled Horizontal Fire Damper 1 X 1	B
0717-005	Ruskin Model NIBD23 3 Hour Type A – U.L. Labeled Horizontal Fire Damper	N01
00735-001	Ruskin Model NIBD23 3 Hour Type C – U.L. Labeled Horizontal Fire Damper 1 X 1	0
2006 Sheet 1	Flow Diagram – Circulating, Screen Wash and Service Water Systems	N76
2031 Sheet 2	Flow Diagram - Reactor Building – Closed Cooling Water System	N65
2036 Sheet 1	Flow Diagram - Reactor Building – Service Water System	N98
2038 Sheet 1	Flow Diagram, Reactor Buiding Floor & Roof Drain Systems	N49
2038 Sheet 2	Flow Diagram, Reactor Buiding Floor & Roof Drain Systems	N03
2040 Sheet 1	Flow Diagram – Residual Heat Removal System	N80
2042	Flow Diagram - Reactor Building – Main Steam System	N85
2045 Sheet 1	Flow Diagram – Core Spray System	N58
2016 Sheet 1C	Flow Diagram – Fire Protection – Reactor Building	N03
2016 Sheet 2	Fire Protection System - Flow Diagram For Pumphouse and Storage Tanks	N30
2016 Sheet 4	Halon and Cardox System Flow Diagram	N04
2041	Reactor Building-Main Steam System-Cooper Nuclear Station	N23
2629-1	8" MS-1 & 10" MS-1 Main Steam	N17
3002 Sheet 1	Auxiliary One Line Diagram Motor Control Center Z, Switchgear Bus 1A, 1B, 1E, And Critical Switchgear Bus 1F, And 1G	N44

3004 Sheet 3	Auxiliary One Line Diagram Motor Control Center C, D, H, J, DG1, And DG2	N22
3012 Sheet 1	Main Three Line Diagram	N08
3012 Sheet 2	Main Three Line Diagram	N06
3012 Sheet 3	Main Three Line Diagram	N19
3012 Sheet 4	Main Three Line Diagram	N13
3012 Sheet 5	Main Three Line Diagram	N15
3012 Sheet 6	Main Three Line Diagram	N17
3012 Sheet 7	Main Three Line Diagram	N08
3012 Sheet 8	Main Three Line Diagram	N07
3012 Sheet 8a	Main Three Line Diagram	N05
3012 Sheet 9	Main Three Line Diagram	N09
3012 Sheet 10	Main Three Line Diagram	N11
3012 Sheet 12	Electrode Boiler Switchgear Main Three Line Diagram	N03
3019 Sheet 3	4160V Switchgear Elementary Diagrams	N36
3020 Sheet 4	4160V Switchgear Elementary Diagrams	N20
3020 Sheet 8	4160V Switchgear Elementary Diagrams	N32
3020 Sheet 9	4160V Switchgear Elementary Diagrams	N22
3020 Sheet 4	4160V Switchgear Elementary Diagrams	N20
3024 Sheet 8	4160V Switchgear Elementary Diagrams Lighting Plan	N32
3045 Sheet 14	Control Elementary Diagrams	N48
3058	D.C. One Line Diagram	N53
3058 Sheet 1	D.C. One Line Diagram	N53
3059, Sheet 1	D.C. Panel Schedules Cooper Nuclear Station	36
3065 Sheet 17	Control Elementary Diagrams	N44
3065 Sheet 17a	Control Elementary Diagram	N11
3177	Outdoor Grounding Plans And Details	N02
3251 Sheet 11	4160V Switchgear Connection Wiring Diagram	N20
3253 Sheet R-1	480V Motor Control Center R Connection Wiring Diagram	N15

3257, Sheet 71	Alternative Shutdown ADS Panel Internal Connections	N06
3700 Sheet 16	Annunciator Elementary Ladder Diagram	N05
3720 Sheet 1	Multiplexer Input Wiring ANN-MUX-10	N04
3726 Sheet 1	Multiplexer Input Wiring ANN-MUX-16	N03
3727 Sheet 1	Multiplexer Input Wiring ANN-MUX-17	N05
3751 Sheet 7	Annunciator Loop Diagram ANN-MUX-01 Devices Sheet No. 6B	N00
3757 Sheet 1	Annunciator Loop Diagram ANN-MUX-07	N01
3766 Sheet 1	Annunciator Loop Diagram ANN-MUX-16	N02
3767 Sheet 1	Annunciator Loop Diagram ANN-MUX-17	N04
0133C8690 Sheet 15	Horizontal Drawout M/C Switchgear Device And Harness Identification	1-17-1973
0223R0558 Sheet 32	Power And Control Circuits Line-Up 08 Units 1 And 2	N22
453200226	Piping Isometric – Wet Sprinkler System Electrical Trays In North East Corner Reactor Building – Floor Elevation 903'-6"	N04
454016108	Contract E69-20 Fire Protection System	N10
454016113	Contract E69-20 Fire Protection System	N01
454016115	Contract E69-20 Fire Protection System	N01
454016116	Contract E69-20 Fire Protection System	N04
454016126	Nebraska Public Power District Contract Number E-69-20	N04
115D6011, Sheet 1	Local Rack 25-50	N00
729E720BB	High Pressure Coolant Injection System	N03
730E149BB, Sheet 1	Functional Control Diagram	N05
730E149BB, Sheet 2	Main Steam Line Isolation Valve Control System Logic	N04
791E253 Sheet 1	Automatic Blowdown System Elementary Diagram	N30
791E253 Sheet 2	Automatic Blowdown System Elementary Diagram	N27
791E253 Sheet 3	Automatic Blowdown System Elementary Diagram	N11
791E264 Sheet 7	Elementary Diagram Reactor Core Isolation Cooling System (13-113)	N15
791E271, Sheet 6	Cooper Nuclear Station-HPCI System-Elementary Diagram	N19

791E266 Sheet 12	Elementary Diagram Primary Containment Isolation System (16-23)	N12
791E514 Sheet 1	Connection Diagram Panel 9-21	N23
791E514 Sheet 2	Connection Diagram Panel 9-21	N01
944E689 Sheet 1	Elementary Diagram (Mod) Low-Low Set	N13
CNS-EQ-105 Sheet 1	EQ Configuration Detail GE/PCI Pressure Switch	N01
CNS-EQ-105 Sheet 2	EQ Configuration Detail, GE/PCI Pressure Switch Tabulation Sheet	N01
CNS-FP-146	932'-6" Reactor Building – North Wall Critical Switchgear Room 1G Fire area Boundary Drawing	N06
CNS-FP-170	Fire Area Boundary Drawing Diesel Generator Room "1" South Wall	N05
CNS-FP-171	Fire Area Boundary Drawing Diesel Generator Room "2" North Wall	N05
CNS-FP-215	Fire Protection Pre-Fire Plan Reactor Building First Floor Elevation 903'-6"	N04
CNS-FP-216	Fire Protection Pre-Fire Plan Reactor Building Critical Switchgear Room 1F Elevation 932'-6"	N03
CNS-FP-221	Fire Protection Pre-Fire Plan Reactor Building MG Set Area Elevation 976'-0"	N05
CNS-FP-236	Fire Protection Pre-Fire Plan Diesel Generator Building D.G. # 1 Elevations 917'-6" and 903'-6"	N05
CNS-FP-285 Sheet 1	CNS Fire Barrier Penetration Seal Details	N04
CNS-EE-186	Safe Shut Down Component Locations & Emergency Route Lighting, 903'-6" Diesel Generator Building	4
CNS-LRP-3, Sheet 4	Local Rack 25-50 Structure	N00
CNS-LRP-3, Sheet 8	Local Rack 25-50 Structure	N01
CNS-LRP-3, Sheet 9	Local Rack 25-50 Structure	N02
E0223R0558, Sheet 33	Power And Control Circuits Line-Up 09 Units 1 And 2 Lighting Plan Sheet 2	N23
E501 Sheet 17A	Integrated Control Circuit Diagram CS-MOV-MO12A Core Spray Inboard Injection Valve	N01
E501 Sheet 17B	Integrated Control Circuit Diagram RHR-MOV-MO25A	N02
E501 Sheet 17C	Integrated Control Circuit Diagram RHR-MOV-MO27A RHR Loop A Injection Outboard Isolation	N02
E501 Sheet 23A	Integrated Control Circuit Diagram RHR-MOV-MO18 RHR Suction Cooling Inboard Isolation Valve	N01

E501 Sheet 26A	Integrated Control Circuit Diagram SW-MOV-M089A RHR Heat Exchanger A Service Water Outlet	N01
E501 Sheet 29C	Integrated Control Circuit Diagram RCIC-MOV-MO21 RCIC Injection	N01
E501 Sheet 30	Motor Operated Valves Connection Diagrams	N08
E501 SHEET30C	Integrated Control Circuit Diagram RHR-MOV-MO17 RHR Shutdown Cooling Supply Outboard Isolation	N01
E501 Sheet 33A	Integrated Control Circuit Diagram HPCI-MOV-MO58 HPCI Pump Suction From Suppression Pool	N01
E501 Sheet 44	Motor Operated Valves Connection Diagrams	N02
E501 Sheet 45A	Integrated Control Circuit Diagram RHR-MOV-MO25B RHR Loop B Injection Inboard Isolation	N02
E501 Sheet 48A	Integrated Control Circuit Diagram SW-MOV-MO89B RHR Heat Exchanger B Service Water Outlet	N02
E507 Sheet 24	Connection Wiring Diagram Reactor Building	N08
E507 Sheet 29	Connection Wiring Diagrams Reactor Building	N03
E507 Sheet 235	Reactor Building Terminal Box 242 Connection Wiring Diagram	N01
G5-262-743 Sheet 1	Emergency Diesel Generator No.1 Electrical Schematic	N23
G5-262-746 Sheet 2	Emergency Diesel Generator No.1 Electrical Schematic	N18
G5-262-746 Sheet 3	Emergency Diesel Generator No.1 Electrical Schematic	N23
G5-262-746 Sheet 4	Emergency Diesel Generator No.1 Electrical Schematic	N12
G5-262-746 Sheet 5	Emergency Diesel Generator No.1 Internal Wiring Diagram	N19
G5-262-746 Sheet 6	Emergency Diesel Generator No.1 Control Panel Wiring Diagram	N16
X2629-200	MS-1 Main Steam	N06

FIRE IMPAIRMENTS

FP08-01-FP-SD-61A&B	FP10-01-NO APPDX R LIGHT	FP10-01-FP-SD-533 CEILING TILE
FP10-02-FP-HT-3 FLOODED	FP10-01-FC9ASDG100F	FP10-01-EE-LTG-APP R
FP10-02-6.FP.302	FP10-01-COMP RM TILES	FP10-01-FP-PNL-CAS
FP10-01-RW BLDG HORNS	FP10-01-CORE BORES	FP10-01-SWP RM HALON
FP10-01-EE-LTG-R18 BULB FAIL	FP10-02-FP-HT-12 IMPAIRED	FP10-02-FP-HT-15 INACCESSABLE
FP10-01-APPDX R FW OVERFILL	FP10-01-WW FALSE ALRM AHU1	FP10-01-FP APP R

FP10-01-6.1FP.6.01 4705129	FP10-01-6.FP.301 4704985	10-0088

PREVENTIVE MAINTENANCE TASKS

4624836	4624889	4663722	4663770	4712840	4713833

PROCEDURES

Number	Title	Revision
Administrative Procedure 0.5	Conduct of the Condition Report Process	67
Administrative Procedure 0.10	Operating Experience Program	21
Administrative Procedure 0.23	CNS Fire Protection Plan	60
Administrative Procedure 0.39	Hot Work	42
Administrative Procedure 0.39.1	Fire Watches and Fire Impairments	6
Emergency Procedure 5.3ALT-STRATEGY	Alternative Core Cooling Mitigating Strategies	23
Emergency Procedure 5.4FIRE-S/D	Fire Induced Shutdown From Outside Control Room	38
Emergency Procedure 5.4POST-FIRE	Post-Fire Operational Information	36 and 37
Maintenance Procedure 15.EE.302	Appendix R/SBO Lighting Functional Test	20
Maintenance Procedure 7.3.21.7	3M Interam E-5A Fire Wrap Fire Resistive Assembly	12
Non-TS Surveillance Procedure 15.FP.303	Fire Detection System Tri-Annual Test (Group 1)	15
Non-TS Surveillance Procedure 15.FP.652	Critical Switchgear Room Duct Wrap Visual Inspection	2
3.9	ASME OM Code Testing Of Pumps and Valves	25
Surveillance	ADS Manual Valve Circuit Continuity from ASD-ADS	11

Procedure 6.ADS.202	Panel	
Surveillance Procedure 6.CSCS.404	IST Closure Test of HPCI-CV-10CV and RCIC-CV-10CV	7
Surveillance Procedure 6.FP.102	Annual Testing of Fire Pumps	30
Surveillance Procedure 6.FP.203	Fire Damper Assembly Examination (Fire Protection System 18 Month Examination)	0 and 9
Surveillance Procedure 6.FP.301	Operations Power Block Sprinkler System Testing	17
Surveillance Procedure 6.FP.302	Automatic Deluge and Pre-Action Systems Testing	19
Surveillance Procedure 6.FP.304	Fire Detection System Circuitry Operability	7
Surveillance Procedure 6.FP.606	Fire Barrier/Fire Wall Visual Examination	12
Surveillance Procedure 6.HPCI.306	Calibration Procedure for HPCI Pressure Instrumentation	8
Surveillance Procedure 6.HPCI.311	HPCI Turbine Trip and Initiation Logic Functional Test	7
Surveillance Procedure 6.SRV.303	Safety Valve and Relief Valve Position Indication Operability Check And LLS Logic Test	13
Surveillance Procedure 6.1FP.301	Diesel Generator CO2 Operability Teat (DIV 1)	10
Surveillance Procedure 6.1FP.302	Fire Detection System 184 Day Examination	9
Surveillance Procedure 6.1FP.601	High Pressure CO2 Cylinder Examination (DIV 1)	12
Surveillance Procedure 7.3.12.2	Safe Shutdown BBESI Emergency Lighting Unit Examination and Maintenance	14
Surveillance Procedure 15.EE.302	Appendix R/SBO Lighting Functional Test	20
Surveillance Procedure 15.FP.305	Fire Detection System Tri-Annual Test (Group 3)	10
System Operating	Communication Systems	41

MISCELLANEOUS DOCUMENTS

Number	Title	Revision
COR002-18-02	OPS-Reactor Core Isolation Cooling	17
Cutler-Hammer	Instructions For Size 1 Or 2 Type B Thermal Overload Relay, 3 Pole, Ambient Compensated Or Non-Compensated I.L.16954A	June 1998
Design Criteria Document 11	Fire Protection Systems	May 10, 2010
Engineering Evaluation Number EE 09-031	Evaluation of Critical Switchgear Rooms 1F and 1G Fire Barrier Separation	0
Evaluation Number EE 04-046	Appendix R MOV Overthrust Evaluation	0
Engineering Procedure Number E-510	Ruskin Manufacturing Company - Site Storage and Handling of NIED-23 Curtain Type Fire Dampers	2
EQDP.2.210	Electroswitch Series 24 (3 Sheets On EQ Certification of Model 24210B Switch)	10
Letter LQA8200158	Fire Protection Rule 10 CFR 50, Appendix R	June 28, 1982
Letter LQA8300109	Fire Protection Rule 10 CFR 50, Appendix R, Preliminary Supplemental Response (Revised)	March 18, 1983
Nebraska Public Power District Letter	Response to Appendix A to Branch Technical Position APCB 9.5-1 Guidelines for Fire Protection for Nuclear Power Plants	December 17, 1976
Nebraska Public Power District Letter	Revisions and Additional Information Fire Protection Review	April 6, 1977
Nebraska Public Power District Letter	Fire Protection Rule 10 CFR 50, Appendix R, Preliminary Supplemental Response (Revision 2)	June 02, 1983
NRC Letter	K. R. Goller, NRC, to Nebraska Public Power District	November 29, 1977
NRC Letter	G. Lear, NRC, to Nebraska Public Power District	February 24, 1978
NRC Letter	T. Ippolito, NRC, to Nebraska Public Power District	May 23, 1979
NRC Letter	T. Ippolito, NRC, to Nebraska Public Power District	September 18, 1979

NRC Letter	T. Ippolito, NRC, to Nebraska Public Power District	November 21, 1980
NRC Letter	D. Vassallo, NRC, to Nebraska Public Power District	April 29, 1983
NRC Letter	D. Vassallo, NRC, to Nebraska Public Power District	September 21, 1983
NRC Letter	D. Eisenhut, NRC, to Nebraska Public Power District	September 21, 1983
NRC Letter	Safety Evaluation For Appendix R to 10 CFR Part 50, Items II.G.3 and III.L, Alternative or Dedicated Shutdown Capability	April 16, 1984
NRC Letter	Outstanding Fire Protection Modifications	August 21, 1985
NRC Letter	W. Long, NRC, to Nebraska Public Power District	April 10, 1986
NRC Letter	W. Long, NRC, to Nebraska Public Power District	September 9, 1986
NRC Letter	Cooper Nuclear Station – Amendment No. 126 to Facility Operation License No. DPR-46	November 7, 1988
NRC Letter	Cooper Nuclear Station – Amendment No. 127 to Facility Operation License No. DPR-46	February 3, 1989
NRC Letter	Revocation Of Exemption From 10 CFR Part 50, Appendix R – Cooper Nuclear Station	August 15, 1995
NRC Letter	Conversion To Improved Technical Specifications For The Cooper Nuclear Station - Amendment No. 178 To Facility Operating License No. DPR-46	July 31, 1998
OTH015-92-02	Lesson Plan Post Fire Shutdown Outside The Control Room Procedures (5.4POST-FIRE, 5.4FIRE-S/D, 5.1ASD)	09
Siemens-Allis DC Contactors	DC Contactors Special Purpose 2 Pole, 600V Max AC or DC Operated Pages 147 And 148	No Date
Siemens Overload 2 Sheets	Manufactures Data Thermal Overload Relays Type 3UA59	April 1997
Siemens Overload 4 Sheets	Manufacture's Data On Bimetallic Thermally Delayed Overload Relays Type 3UA5, 3UA6 Class 10	No Date
Southwest Research Institute	NPPD PO# 4500092806 Williams Fire Pump Diesel Oil Test Summary Report	July 29,2008
Southwest Research Institute	NPPD PO# 4500100440 Williams Fire Pump Diesel Oil Analytical Test Report	Revision 1 May 11, 2009
Southwest Research Institute	NPPD PO# 4500102145 Williams Fire Pump Diesel Oil Analytical Test Report	May 18, 2010
Technical Publication	Electroswitch Series 24 Instrument and Control	February 1998

24-1	Switches For Power Industry and Heavy Duty Industrial Applications	
Technical Requirements Manual Section 3.11	Fire Protection Systems	July 29, 2010
Technical Specification 3.3.3.2	Alternative Shutdown System	Amendment 233
Updated Safety Analysis Report Section VII-18	Alternative Shutdown Capability	July 24, 2001
Updated Safety Analysis Report Section X-9	Fire Protection System	January 08. 2004
Updated Safety Analysis Report Section X-18	Appendix R Safe Shutdown	January 29, 2003
Updated Safety Analysis Report Section XIII-10	Fire Protection Program	April 16, 2010
VM-1730	Emergency Lighting	1
Westinghouse Starter Information	Manufactures Data Sheets Showing 460 VAC A201, A211, A251 Size 2 Magnetic Contactor Non-Reversing Or Reversing I.L. 16961A	April 1984
257HA354AC	GE Design Specification, Sheet 2	2
790523	Amendment No. 56 to Facility Operating License No. DPR – 46	001
4605196	Sample Fuel Oil And Send For Analysis For Williams B.5.b Credited Pump	July 29, 2008
4625867	Sample Fuel Oil And Send For Analysis For Williams B.5.b Credited Pump	April 29, 2009
4664953	Sample Fuel Oil And Send For Analysis For Williams B.5.b Credited Pump	May 03, 2010
	IST Reference/Acceptance Limits Data File	205

SYSTEM TRAINING MANUALS

Number	Title	Revision
COR002-11-02	High Pressure Coolant Injection	26
COR002-19-02	Reactor Equipment Cooling	20
COR002-23-02	Residual Heat Removal System	27

COR002-34-02	Alternative Shutdown System	18

WORK ORDERS

4704976	4704973	4705129	4636801	4704980	4705274	4704985	4704986
4705369	4541652	4680341	4600849	4601469	4625865	4627329	4629553
4634534	4636434	4643635	4648115	4649842	4656140	4659221	4659685
4662049	4664951	4688234	4691445	4694802	4702636	4704770	4711699
4712867	4713861						

FINAL SIGNIFICANCE DETERMINATION SUMMARY
COOPER TRIENNIAL FIRE PROTECTION ISSUE

Significance Determination Basis

a. Phase 1 Screening Logic, Results, and Assumptions

In accordance with NRC Inspection Manual Chapter 0612, Appendix B, "Issue Screening," the issue was determined to be more than minor because it was associated with the equipment performance attribute and affected the mitigating systems cornerstone objective to ensure the availability, reliability, or function of a system or train in a mitigating system in that 3 motor-operated valves would not have functioned following a postulated fire in multiple fire zones. The following summarizes the valves and fire areas affected:

- Valves Affected

RHR-MO-25A Residual Heat Removal (RHR) A Inboard Injection Valve
RHR-MO-25B RHR B Inboard Injection Valve
RR-MO-53A Reactor Recirculation Pump A Discharge Valve

- Fire Areas Affected

CB-A-1	Control Building Division 1 Switchgear Room and Battery Room
CB-B	Control Building Division 2 Switchgear Room and Battery Room
CB-C	Control Building Reactor Protection System Room 1B
CB-D	Control Room, Cable Spreading Room, Cable Expansion Room, and Auxiliary Relay Room
RB-DI (SW)	Reactor Building South/Southwest 903, Southwest Quad 889 and 859, and RHR Heat Exchanger Room B
RB-DI (SE)	Reactor Building RHR Pump B/HPCI Pump Room
RB-J	Reactor Building Critical Switchgear Room 1F RB-K Reactor Building Critical Switchgear Room 1G
RB-M	Reactor Building North/Northwest 931 and RHR Heat ExchangerRoom
RB-N	Reactor Building South/Southwest 931 and RHR Heat Exchanger Room B
TB-A	Turbine Building (multiple areas)

The significance determination process (SDP) Phase 1 Screening Worksheet (Manual Chapter 0609, Attachment 4), Table 3b directs the user to Manual Chapter 0609, Appendix F, "Fire Protection Significance Determination Process," because it affected fire protection defense-in-depth strategies involving post fire safe shutdown systems. However, Manual Chapter 0308, Attachment 3, Appendix F, "Technical Basis for Fire Protection Significance Determination Process for at Power Operations," states that Manual Chapter 0609, Appendix F, does not include explicit treatment of fires in the main control room. The Phase 2 process can be utilized in the treatment of main control room fires, but it is recommended that additional guidance be sought in the conduct of such an analysis.

b. Phase 2 Risk Estimation

Based on the complexity and scope of the subject finding and the significance of the finding to main control room fires, the analyst determined that a Phase 2 estimation was not appropriate.

c. Phase 3 Analysis

A risk analysis was performed previously of a similar problem that affected the three valves addressed by this performance deficiency. This was documented in EA 07-204, Report Number 05000298/2008008, dated June 13, 2008. In both cases, Valves RHR-MOV-25A, RHR-MOV-25B, and RHR-MOV-53A were incapable of being remotely operated from the motor starter as prescribed by Procedure 5.4FIRE-S/D. The risk estimate performed in 2008 as it pertains to these three valves (the 2008 Phase 3 also included several other valves) remains valid for the current situation. However, changes were made to Procedure 5.4FIRE-S/D subsequent to the 2008 issue. These changes were credited in the current analysis and resulted in a decrease in the risk significance of the subject valves. Text from the 2008 risk analysis is shown in italics throughout this document.

In accordance with Manual Chapter 0609, Appendix A, the analyst performed a Phase 3 analysis using input from the Nebraska Public Power District, "Individual Plant Examination for External Events (IPEEE) Report – 10 CFR 50.54(f) Cooper Nuclear Station, NRC Docket No. 50-298, License No. DPR-46," dated October 30, 1996, the Standardized Plant Analysis Risk (SPAR) Model for Cooper, Revision 3.31, dated September 2007, licensee input (see documents reviewed list in Enclosure 3), a probabilistic risk assessment using a linked event tree model created by the analyst for evaluating main control room evacuation scenarios, and appropriate hand calculations. **[Note: The SPAR model used in the 2008 analysis has been superseded by newer versions. However, the risk result gained from the portion of the analysis that used this model (non-alternative shutdown scenarios) was not significant to the current risk estimate. Virtually all of the risk associated with the current issue results from the alternative shutdown scenarios for which a specific SPAR model was created. Therefore, the use of the older model has no consequence.]**

<u>Assumptions</u>:

1. *For fire zones that do not have the possibility for a fire to require the main control room to be abandoned, the ignition frequency identified in the IPEEE is an appropriate value.*

2. *The fire ignition frequency for the main control room (P_{FIF}) is best quantified by the licensee's revised value of 6.88 x 10^{-3}/yr.*

3. *Of the original 64 fire scenarios evaluated, 18 were determined to be redundant and were eliminated, 41 of the remaining (documented in Table 1) were identified as the predominant sequences associated with fires that did not result in control room abandonment.* **[Note: the current issue did not include all of the fire scenarios from the 2008 issue, but all of the current fire scenarios are included in the 2008 compilation]**

4. *The baseline conditional core damage probability for a control room evacuation at the Cooper Nuclear Station is best represented by the creation of a probabilistic risk assessment tool previously created by the analyst using a linked event tree method. The primary event tree used in this model is displayed as Figure 1 in the Attachment. The baseline conditional core damage probability as calculated by the linked event tree model was 1.14×10^{-1}, which is similar to the generic industry value of 0.1.*

5. *The analyst used an event tree, RECOVERY-PATH, shown in Figure 2 in the Attachment, to evaluate the likelihood of operator recovery via either restoration of HPCI or manually opening Valve RHR-MO-25B. The resulting non-recovery probability was 7.9×10^{-2}.* **[Note: This value was adjusted to 1.01E-3 in the current analysis based on improvements made to Procedure 5.4FIRE-S/D.]**

6. *The risk related to a failure of Valve RHR-MO-25B to open following an evacuation of the main control room was evaluated using the analyst's linked event tree model. The conditional core damage probability calculated by the linked event tree model was 1.19×10^{-1}.*

7. *Any fire in the main control room that is large enough to grow and that goes unsuppressed for 20 minutes will lead to a control room evacuation.*

8. *Any fire that is unsuppressed by automatic or manual means in the auxiliary relay room, the cable spreading room, the cable expansion room or Area RB-FN will result in a main control room evacuation.*

9. *The Cooper SPAR model, Revision 3.31, represents an appropriate tool for evaluation of the core damage probabilities associated with postulated fires that do not result in main control room evacuation.*

10. *All postulated fires in this analysis resulted in a reactor scram. In addition, the postulated fire in Fire Area RB-K resulted in a loss-of-offsite power.*

11. *Valves RHR-MO-25A and RHR-MO-25B are low pressure coolant injection system isolation valves. These valves can prevent one method of decay heat removal in the shutdown cooling mode of operation.*

12. *For Valves RHR-MO-25A and RHR-MO-25B, the subject performance deficiency only applies to the portion of the post fire procedures that direct the transition into shutdown cooling.*

13. *Valve RHR-MO-25B must opened from the motor-control center for operators to initiate alternative shutdown cooling from the alternative shutdown panel following a main control room evacuation.*

14. *Valve RHR-MO-53A is the discharge isolation valve for Reactor Recirculation Pump 1-A. The failure to close either this valve or Valve RR-MO-43A would result in a short circuit of the shutdown cooling flow to the reactor vessel. The performance deficiency did not apply to Valve RR-MO-43A.*

15. *The exposure time used for evaluating this finding should be determined in accordance with Inspection Manual Chapter 0609, Appendix A, Attachment 2, "Site Specific Risk-Informed Inspection Notebook Usage Rules." Given that the performance deficiency was known to have existed for many years, the analyst used the 1-year of the current assessment cycle as the exposure period.*

16. *Based on fire damage and/or procedures, equipment affected by a postulated fire in a given fire zone is unavailable for use as safe shutdown equipment.*

17. *The performance deficiency would have resulted in each of the demanded valves failing to respond following a postulated fire.*

18. *In accordance with the requirements of Procedure 5.4POST-FIRE, operators would perform the post-fire actions directed by the procedure following a fire in an applicable fire zone. Therefore, the size and duration of the fire would not be relevant to the failures caused by the performance deficiency.*

19. *Given Assumption 18, severity factors and probabilities of non-suppression were not addressed for postulated fires that did not result in main control room evacuation.*

<u>Postulated Fires Not Involving Main Control Room Evacuation</u>:

The risk significance from fires not involving control room evacuation was determined to be insignificant for the current finding. This was estimated by referring to the 2008 risk evaluation. Text in italics is from the 2008 report and Table 1 is reproduced for the fire areas that involve RHR-MOV-25A, RHR-MOV-25B, or RHR-MOV-53A.

The senior reactor analyst used the SPAR model for Cooper Nuclear Station to estimate the change in risk, associated with fires in each of the associated fire scenarios (Table 1, Items 1 – 41) that was caused by the finding. Average unavailability for test and maintenance of modeled equipment was assumed, and a cutset truncation of 1.0×10^{-13} was used. For each fire zone, the analyst calculated a baseline conditional core damage probability consistent with Assumptions 9, 10, 25 **[now 17]** *and 26* **[now 18]**.

For areas where the postulated fire resulted in a reactor scram, the frequency of the transient initiator, IE-TRANS, was set to 1.0. All other initiators were set to the house event "FALSE," indicating that these events would not occur at the same time as a reactor scram. Likewise, for Fire Area RB-K, the frequency of the loss-of-offsite power initiator, IE-LOOP, was set to 1.0 while other initiators were set to the house event "FALSE."

With input from the detailed IPEEE notebooks, maintained by the licensee, the analyst was able to better assess the fire damage in each zone. This resulted in a more realistic evaluation of the baseline fire risk for the zone, and lowering the change in risk for each example.

Consistent with guidance in the Reactor Accident Sequence Precursor Handbook, including NRC document, "Common-Cause Failure Analysis in Event Assessment, (June 2007)," the baseline established for the fire zone, and Assumptions 22 through 26, **[now 15 through 19]** *the analyst modeled the resulting condition following a postulated fire in each fire zone by adjusting the appropriate basic events in the SPAR model. Both the baseline and conditional values for each fire zone are documented in Table 1.*

As shown in Table 1, the analyst calculated a change in core damage frequency (ΔCDF) associated with these 41 fire scenarios of 2.9 x 10^{-6}*/yr.* **[Note: This result included fire areas not affected by the current finding.]**

The analyst evaluated the licensee's qualitative reviews of the 13 fire scenarios that were impacted by the failure of the HPCI turbine to trip. In these scenarios, HPCI floods the steam lines and prevents further injection by either HPCI or reactor core isolation cooling system. Qualitatively, not all fires will grow to a size that causes a loss of the trip function due to spatial separation. Additionally, not all unsuppressed fires would cause a failure of the HPCI trip function. Finally, no operator recovery was credited in these evaluations.

Given that these qualitative factors would all tend to decrease the significance of the finding, the analyst believed that the total change in risk would be significantly lower than the 2.9 x 10^{-6}*/yr documented above. Based on analyst judgment and an assessment of the evidence provided by the licensee, an occurrence factor of 0.1 was applied to the13 fire scenarios. This resulted in a total ΔCDF of 7.8 x* 10^{-7}*/yr. Therefore, the analyst determined that this value was the best estimate of the safety significance for these 41 fire scenarios.*

From Table 1, the total risk associated with fire areas that involve Valves RHR-MOV-25A, RHR-MOV-25B, or RHR-MOV-53A is 5.5E-7. As noted above, in the 2008 analysis, there were qualitative reasons for lowering this risk estimate. Also, because the previous evaluation included the contribution from several other valves that affected the same fire areas, the risk attributable to the current evaluation is lower. For these reasons, the analyst concluded that the risk for the current finding is less than 1.0E-7 for fire areas that do not involve control room evacuation.

TABLE 1
Postulated Fires Not Involving Main Control Room Evacuation

Fire Area/Shutdown Strategy	Area/-Zone	Scenario Number	Scenario Description	Ignition Frequency	Base CCDP	Case CCDP	Estimated delta-CDF Contribution	Function Affected
RBC-CF	1C	1	RHR A Pump Room	2.94E-03	8.82E-07	8.15E-05	2.37E.07	
		2	MCC K	3.02E-03	2.76E-05	1.28E-04	3.03E-07	
		3	MCC Q	3.93E-03	2.76E-05	1.28E-04	3.95E-07	
		4	MCC R	3.43E-03	2.76E-05	1.28E-04	3.44E-07	
		5	MCC RB	1.62E-03	1.12E-03	1.21E-03	1.46E-07	
		6	MCC S	2.23E-03	1.12E-03	1.21E-03	2.01E-07	Shut HPCI-MO-14,
		7	MCC Y	3.83E-03	1.12E-03	1.21E-03	3.45E-07	HPCI-MO-16,
		8	Panel AA3	9.98E-04	2.76e-05	1.28E-04	1.00E-07	RHR-MO-921,
	2A/2C	9	Panel BB3	9.98E-04	1.12E-03	1.21E-03	8.98E-08	RWCU-MO-18 and
		10	RCIC Starter Rack	1.32E-03	5.27E-06	8.27E-05	1.02E-07	MS-MO-77
		11	250V Div 1 Rack	5.10E-04	2.76E-05	1.28E-04	5.12E-08	
		12	250V Div 2 Rack	2.09E-04	1.12E-03	1.21E-03	1.88E-08	
		13	ASD Panels	3.02E-04	1.12E-03	1.21E-03	2.72E-08	
CB-A		14		6.74E-03	7.64E-04	7.64E-04	0.00E+00	
		15		1.36E-03	2.61E-06	2.61E-06	0.00E+00	
		16	RPS Room 1A	4.15E-03	1.75E-07	1.75E-07	0.00E+00	Open RHR-MO-25B
		17		2.42E03	3.57E-04	3.58E-04	4.84E-10	and RHR-MO-67
		18	Hallway (used CB corridor)	1.09E-02	2.05E-05	2.85E-05	8.74E-08	

Fire Area/Shutdown Strategy	Area/-Zone	Scenario Number	Scenario Description	Ignition Frequency	Base CCDP	Case CCDP	Estimated delta-CDF Contribution	Function Affected
CB-A1	8H	19	DC Switchgear Room 1A	4.27E-03	3.49E-03	3.49E-04	1.28E-09	Open RHR-MO-17, RHR-MO-25B, and RHR-MO-67
	8E	20	Battery Room 1A	2.25E-03	8.74E-06	1.03E-05	3.51E-09	
CB-B	8G	21	DC Switchgear Room 1B	4.27E-03	1.82E-03	1.83E-03	3.42E-08	Open RHR-MO-25A
	8F	22	Battery Room 1B	2.25E-03	4.81E-06	5.73E-06	2.07E-09	
CB-C	8B	23	RPS Room 1A	4.15E-03	1.75E-07	1.77E-07	5.81E-12	Open RHR-MO-17, RHR-MO-25A, and RHR-MO-67
	8C	24		4.15E-03	1.75E-07	1.77E-07	5.81E-12	
RB-DI (SW)	2D	25	RHR Heat Exchanger Room B	6.70E-04	8.66E-05	8.68E-05	1.27E-10	Shut HPCI-MO-14 and RR-MO-53A
RB-DI (SE)	1D/1E	26	RHR B/HPCI Pump Room	4.28E-03	6.48E-05	1.44E-04	3.37E-07	Shut HPCI-MO-14 and RR-MO-53A
RB-J	3A	27	Switchgear Room 1F	3.71E-03	5.28E-05	5.28E-05	0.00E+00	Open RHR-MO-17, RHR-MO-25B, and RHR-MO-67
RB-L	3B	28	Switchgear Room 1G	3.71E-03	1.77E-02	1.77E-02	0.00E+00	Open RHR-MO-25A
RB-M	3C/3D/3E	29	RB Elevation 932	1.13E-02	7.06E-06	8.99E-06	2.18E.08	Open RHR-MO-17 and RHR-MO-25B
	2B	30	RHR Hx Room A	6.70E-04	7.06E-06	8.99E-06	1.29E-09	
RB-N	3C/3D/3E	31	Reactor Building Elevation 932	1.13E-02	1.22E-05	1.38E-05	1.81E-08	Open RHR-MO-25A
	2D	32	RHR Heat Exchanger Room B	6.70E-04	1.22E-05	1.38E-05	1.07E-09	

Fire Area/Shutdown Strategy	Area/-Zone	Scenario Number	Scenario Description	Ignition Frequency	Base CCDP	Case CCDP	Estimated delta-CDF Contribution	Function Affected
TB-A	11D	33	Condenser Pit Area	3.10E-03	4.83E-06	6.20E-06	4.25E-09	Open RHR-MO17, RHR-MO-25A, and RHR-MO-67
	11E	34	Reactor Feedwater Pump Area	6.25E-03	4.83E-06	6.20E-06	8.56E-09	
	11L	35	Pipe Chase	6.70E-04	4.83E-06	6.20E-06	9.18E-10	
	12C	36	Condenser and Heater Bay Area	3.27E-03	4.83E-06	6.20E-06	4.48E-09	
	12D	37	TB Floor 9033	3.45E-03	4.83E-06	6.20E-06	4.73E-09	
	13A	38	Operating Floor Non-critical	5.76E-03	4.83E-06	6.20E-06	7.89E-09	
	13B	39	Switchgear Room	3.79E-03	4.83E-06	6.20E-06	5.19E-09	
	13C	40	Electric Shop	8.56E-04	4.83E-06	6.20E-06	1.17E-09	
	13D	41	I&C Shop	8.90E-04	4.83E-06	6.20E-06	1.22E-09	
Total Estimated △CDF for 41 Postulated Fire Scenarios							2.91E-06	

Post-Fire Remote Shutdown Calculations:

Note: The risk attributable to post-fire remote shutdown (control room abandonment sequences) results predominantly from the inability to operate Valve RHR-MOV-25B as described in Procedure 5.4FIRE-S/D. This is the credited train and the only procedural means for initiating shutdown cooling during the recovery actions. The additional risk contribution from RHR-MOV-25A and RHR-MOV-53A is negligible.

As documented in Assumptions 4, 5, and 6, the analyst created a linked event tree model, using the Systems Analysis Programs for Hand-on Integrated Reliability Evaluation (SAPHIRE) software provided by the Idaho National Laboratory, to evaluate the risks related to fire-induced main control room abandonment at the Cooper Nuclear Station. This linked event tree was used to evaluate the increased risk from the subject performance deficiency during the response to postulated fires in the main control room, the auxiliary relay room, the cable spreading room, the cable expansion room or Fire Area RB-FN. The primary event tree used in this model is displayed as Figure 1 in the Attachment.

As documented in Assumption 5, the analyst used an event tree to evaluate the likelihood of operator recovery via either restoration of HPCI or manually opening Valve RHR-MO-25B. The resulting non-recovery probability was 1.01E-3. The derivation of this result is discussed below. This result applied only to sequences where HPCI provides injection flow. In cases where HPCI fails or is not available, there is much less time available to recover from the failure. For this case, a SPAR-H evaluation was performed, and is discussed below.

Note: In the 2008 analysis, the non-recovery probability for HPCI success sequences was determined to be 7.9E-2. This non-recovery probability was decreased by a factor of 78 for the current finding because of changes that were made to Procedure 5.4FIRE-S/D. These changes directed operators to close SRVs if RHR injection was not observed to be successful. Also, it directed operators to delay securing HPCI until RHR injection is confirmed.

In the 2008 analysis, recovery credit was only applied to sequences that contained an early success (lack of failure or unavailability) of HPCI. This is because with the use of HPCI, a considerable amount of decay heat is removed prior to the point of attempting to open RHR-MOV-25B in Procedure 5.4FIRE-S/D, and ample time is available to diagnose the failure and manually open the valve prior to fuel damage. Also, HPCI can be re-initiated in these cases to maintain reactor parameters, and the new procedures instruct operators to keep HPCI online until low-pressure injection is confirmed. However, if HPCI is out of service for maintenance or experiences a failure, the only success path is to establish RHR low pressure injection and the time available is very limited. According to the licensee's MAAP analysis, incipient core damage will occur 15 minutes after RHR-MOV-25B fails to open unless it is opened (manually) by that time. For early HPCI failures, it is assumed in this analysis (consistent with the 2008 analysis) that there is enough time to reach the step in Procedure 5.4FIRE-S/D where RHR-MOV-25B is opened. If it fails to open (1.2E-2 in the base case, 1.0 in the condition case), operators have 15 minutes to diagnose the situation (injection failure) and develop a strategy that includes visually checking the position of RHR-MOV-25B and opening it manually to at least 23 hand wheel turns to get sufficient flow to prevent core damage.

The analyst considered whether changes to Procedure 5.4FIRE-S/D subsequent to the 2008 risk analysis could allow some recovery credit to be applied to sequences involving early HPCI failure in the current analysis. One possible reason to do this is that the revised procedure directs the operator at the alternative shutdown panel to close SRVs in the event that RHR injection cannot be verified. This would have the effect of delaying the depletion of water inventory in the core. However, the diagnosis of this situation would likely take a long time. The operator at the alternative shutdown panel would be difficult to determine quickly, whether low pressure injection was successful because of a lack of direct indication (total RHR flow is displayed, but the effect of successful injection would only be a slight increase in the total RHR flow rate until Valve RHR-MO-34B is throttled closed to divert the flow that was previously directed to the suppression pool). The reactor level indication would likely be the first indication of unsuccessful injection, but a lowering level could well be misinterpreted as a shrink from the injection of colder water. Also, if the operator used the alternative method prescribed in the procedure, which is used when nitrogen pressure is determined to be reliably available, he is directed to use SRVs to maintain pressure within a band of 150-200 psig. This could result in masking the lowering level from a lack of injection. For these reasons, the analyst determined that recovery for early HPCI failure sequences would be challenging.

A SPAR-H evaluation was performed to estimate a non-recovery probability for HPCI failure sequences. All non-nominal PSFs are shown in the following table:

	Diagnosis (nominal =1.0E-2)	Action (nominal = 1.0E-3)
Available Time	Barely Adequate (2/3 nominal) (10)	Time Required (10)
Stress	High (2)	High (2)
Complexity	Moderate (2)	Nominal
Experience/Training	Nominal	High (0.5)
Procedures	Poor (5)	Nominal
Ergonomics	Nominal	50% Poor, 50% nominal (5.5)
Total PSF Product	200	55
HEP	0.67	0.05
Total HEP	0.72	

The licensee's thermal-hydraulic analysis indicated that approximately 15 minutes of time would be available to open RHR-MOV-25B enough turns to provide adequate core flow after the step in the procedure to open RHR-MOV-25B failed. The analyst assumed that a nominal time to diagnose the problem is 15 minutes and the nominal time to close the valve is 5 minutes. The available 15 minutes was partitioned with 10 minutes for diagnosis and 5 minutes for action. This explains the selection of the factors above for available time for both diagnosis and action.

Stress would be high in both cases. For diagnosis, complexity was considered to be moderate because of the need to observe several indications while following a procedure that only addresses successful operation of the equipment and that directs further actions to be taken that are unrelated to diagnosing equipment failures. In addition, procedures for diagnosis were considered to be poor because of a lack of direction to the operator at the alternative shutdown panel to check the position of RHR-MOV-25B if a reactor vessel rise is not observed. Although there is a procedural step for the reactor building operator to check the valve position, it is specifically prescribed for cable spreading room fires only, and it is not clear that he would do this for other alternative shutdown fires unless directed by the operator at the alternative shutdown panel. The analyst considered experience and training to be high for MOV manual operations at the plant because it is a frequently performed task. Ergonomics for action were divided half and half between poor and nominal because it would take an unusually large force to open the valve against the full shutoff head of the RHR pump. In addition, there is a somewhat unfavorable geometry for this operation.

Procedure 5.4FIRE-S/D, Attachment 2, Step 1.20.7 instructs the reactor building operator to verify that RHR-MOV-25B is open if the fire is in the cable spreading room. If the valve is observed to not be open, Step 1.20.8 instructs the operator to open the breaker and manually open the valve. There is some uncertainty as to whether the operator would proceed with Step 1.20.8 (after correctly skipping Step 1.20.7) if the fire was not in the cable spreading room. The analyst concluded that the text of Step 1.20.8 ("If the valve did not operate, perform following..") is written in such a way that it presumes that the operator has performed the valve position verification of Step 1.20.7. Therefore, if Step 1.20.7 is skipped, it would be logical to mark Step 1.20.8 "N/A."

The analyst concluded that the recovery probability for cable spreading room fires would be nominal because it involves a direct observation of the valve position, followed by a well-trained and proceduralized evolution. Therefore, for cable spreading room fires, the non-recovery probability was assigned a value of 1.1E-2 (nominal SPAR-H value). Unlike the value used for "action" in the SPAR-H tabulation above, in this case there would be extra time available for the operator to open the valve manually because no time would be needed for diagnosis. For all other fire areas that cause alternative shutdown, the non-recovery value of 0.72 was used as discussed above. The following table summarizes the recovery assumptions:

	Non-Recovery Value
HPCI Success	1.01E-3
Early HPCI Failure Cable Spreading Room	1.1E-2
Early HPCI Failure All Other ASD Areas	0.72

Using the linked event tree model described in Assumption 4, the analyst calculated the Condition CDF as 7.79E-6/yr. The base CDF was 5.81E-6/yr. With a one-year exposure time, the delta-CDF is 2.0E-6/yr. Almost all of the risk (approximately 99%) resulted from sequences that involve alternative shutdown fires (other than the cable spreading room) that include early failures or unavailability of HPCI.

The dominant cutsets are shown below in Table 2.

Table 2 Main Control Room Abandonment Sequences			
Postulated Fire	Sequence	Mitigating Functions	Results
Auxiliary Relay Room	**4-01-12**	**Early Failure of HPCI Failure to Open MO-25B**	**1.3 x 10^{-6}/yr**
Main Control Room	**3-01-12**	**Early Failure of HPCI Failure to Open MO-25B**	**3.4 x 10^{-7}/yr**
Auxiliary Relay Room	**4-31-1-1-1-1-12**	**Early Failure of HPCI Failure to Open MO-25B**	**1.8 x 10^{-7}/yr**
Main Control Room	**3-31-1-1-1-1-12**	**Early Failure of HPCI Failure to Open MO-25B**	**4.6 x 10^{-8}/yr**
Auxiliary Relay Room	**4-01-03**	**Early Failure of HPCI Failure to Open MO-25B**	**3.4 x 10^{-8}/yr**

The following text from the 2008 analysis discusses the derivation of the control room abandonment frequency. This information was considered applicable to the current evaluation.

<u>Control Room Abandonment Frequency</u>

NUREG/CR-2258, "Fire Risk Analysis for Nuclear Power Plants," provides that control room evacuation would be required because of thick smoke if a fire went unsuppressed for 20 minutes. Given Assumption 6 and assuming that a fire takes 2 minutes to be detected by automatic detection and/or by the operators, there are 18 minutes remaining in which to suppress the fire prior to main control room evacuation being required. NRC Inspection Manual Chapter 0609, Appendix F, Table 2.7.1, "Non-suppression Probability Values for Manual Fire Fighting Based on Fire Duration (Time to Damage after Detection) and Fire Type Category," provides a manual non-suppression probability (P_{NS}) for the control room of 1.3×10^{-2} given 18 minutes from time of detection until time of equipment damage. This is a reasonable approach, although fire modeling performed by the licensee indicated that 16 minutes was the expected time to abandon the main control room based on habitability.

In accordance with Inspection Manual Chapter 0609, Appendix F, Task 2.3.2, the analyst used a severity factor of 0.1 for determining the probability that a postulated fire would be self sustaining and grow to a size that could affect plant equipment.

Given these values, the analyst calculated the main control room evacuation frequency for fires in the main control room (F_{EVAC}) as follows:

$$F_{EVAC} = P_{FIF} * SF * P_{NS}$$

$$= 6.88 \times 10^{-3}/yr * 0.1 * 1.3 \times 10^{-2}$$

$$= 8.94 \times 10^{-6}/yr$$

In accordance with Procedure 5.4FIRE-S/D, operators are directed to evacuate the main control room and conduct a remote shutdown, if a fire in the main control room or any of the four areas documented in Assumption 8, if plant equipment spuriously actuates/de- energizes equipment, or if instrumentation becomes unreliable.

Therefore, for all scenarios except a postulated fire in the main control room, the probability of non- suppression by automatic or manual means are documented in Table 3, below.

Table 3 Control Room Abandonment Frequency					
Fire Area	Ignition Frequency (per year)	Severity	Automatic Suppression	Manual Suppression	Abandonment Frequency (per year)
Main Control Room	6.88×10^{-3}	0.1	none	1.3×10^{-2}	8.94×10^{-6}
Auxiliary Relay Room	1.42×10^{-3}	0.1	none	0.24	3.41×10^{-5}
Cable Expansion Room	1.69×10^{-4}	0.1	2×10^{-2}	0.24	8.11×10^{-8}
Cable Spreading Room	4.27×10^{-3}	0.1	5×10^{-2}	0.24	5.12×10^{-6}
Reactor Building 903' (RB-FN)	1.43×10^{-3}	0.1	2×10^{-2}	0.24	6.86×10^{-7}
Total MCR Abandonment:					4.89×10^{-5}

The licensee's total control room abandonment frequency was 1.75×10^{-5}. For the main control room fire, the licensee's calculations were more in-depth than the analyst's. The remaining fire areas were assessed by the licensee using IPEEE data. However, the following issues were noted with the licensee's [2008] assessment:

Kitchen fires were not included in licensee's evaluation

- *This would tend to increase the ignition frequency*
- *This might add more heat input than the electrical cabinet fires modeled by the licensee*

Habitability Forced Abandonment

- *Non-suppression probability did not account for fire brigade response time or the expected time to damage.*
- *Reduced risk based on 3 specific cabinets causing a loss of ventilation early, when it should have increased the risk. Fire modeling showed that fires in these cabinets could damage nearby cables and cause ventilation damper(s) to close.*
- *Risk Assessment Calculation ES-91 uses an abandonment value of 9.93×10^{-7}. However, the supporting calculation performed by EPM used 3.02×10^{-6}.*

Equipment Failure Control Room Abandonment

- *Criteria for leaving the control room did not accurately reflect the guidance that was proceduralized.*

- *The evaluation of the Cable Expansion Room stated that the only fire source was self-ignition of cables. This was modeled as a hot work fire, and it included a probability that administrative controls for hot work and fire watches would prevent such fires from getting large enough to require control room abandonment. This is inappropriate for self-ignition of cables, since there would not really be any fire watch present. Adjusting for this would increase the risk in this area by two orders of magnitude.*

- *The licensee concluded that fires in equipment in the four alternative shutdown fire areas outside the main control room (see Assumption 8) would not result in control room abandonment without providing a technical basis. The licensee's Appendix R analysis concluded that fire damage in these rooms require main control room evacuation to prevent core damage.*

The analyst used the main control room abandonment frequencies documented in Table 3. In addition, sensitivities were run using the licensee's values.

<u>Recovery Following Failure of Valve RHR-MO-25B (HPCI success sequences only)</u>

As noted above, the recovery value determined in the 2008 analysis was 7.9E-2. The following table presents the revised split fractions based on the improvements to Procedure 5.4FIRE-S/D.

Table 4
Split Fractions for RECOVERY-PATH

Top Event	How Assessed	Failure Probability
LEVEL-DOWN	SPAR-H (Diagnosis Only)	1.75E-4
SRV-STATUS	SPAR-H (Diagnosis Only)	1.75E-3
CLOSE-SRVS	SPAR-H (Action Only)	4.38E-4
RESTORE-HPCI	SPAR-H (Combined)	7.0E-4
OPEN-MO-25B	SPAR-H (Combined)	2.89E-1

Using the event tree in Figure 2 and the split fractions in Table 4, the analyst calculated a combined non-recovery probability of 1.01E-3.

The licensee's combined non-recovery probability was 4.0×10^{-3}. **[Note: this value is based on the licensee's evaluation before the aforementioned improvements were made to the procedure].** *The licensee used a similar approach to quantify this value. However, the licensee assumed that operators would always shut the safety-relief valves upon determining that reactor pressure vessel water level was decreasing. The analyst assumed that some percentage of operators would continue to follow the procedure and attempt to recover from the failed RHR valve or try alternative methods of low-pressure injection. In addition, the analyst identified the following issues that impacted the licensee's analysis:*

The inspectors determined that it would require 112 ft-lbs of force to manually open Valve RHR-MO-25B. The analyst determined that this affected the ergonomics of this recovery. Some operators may assume that the valve is on the backseat when large forces are required to open it. Some operators might be incapable of applying this force to a 2-foot diameter hand wheel.

The analyst noted that the following valves would be potential reasons for lack of injection flow and/or may distract operators from diagnosis that Valve RHR-MO-025B is closed:

- *RHR-81B, RHR Loop B Injection Shutoff Valve, could be closed.*
- *RHR-27CV, RHR Loop B Injection Line Testable Check Valve, could be stuck closed.*
- *RHR-MO-274B, Injection Line Testable Check Valve Bypass Valve, could be opened as an alternative.*
- *Operators could search for an alternative flow path.*

The licensee's [2008] evaluation did not include sequences involving the failure of the HPCI system shortly after main control room evacuation in their risk evaluation. These sequences represented approximately 26 percent of the ΔCDF as calculated by the analyst. These sequences are important for the following reasons:

- *Failure of HPCI leads to the need for operators to rapidly depressurize the reactor to establish alternative shutdown cooling. Decay heat will be much higher than for sequences involving early HPCI success. Also, depressurization under high decay heat and high temperature result in greater water mass loss. This will significantly reduce the time available for recovery actions.*
- *HPCI success sequences provide long time frames available with HPCI operating. This reduces decay heat, increases time for recovery, and permits the establishment of an emergency response organization. Those factors are not applicable to early HPCI failure sequences.*

The basis for operating HPCI was not well documented by the licensee. During many of the extended sequences, suppression pool temperature went well above the operating limits for HPCI cooling and remained high for extended periods of time. The following facts were determined through inspection:

- *The design temperature for operating HPCI is 140°F based on process flow providing oil cooling.*
- *General Electric provided a transient operating temperature of 170°F for up to 2 hours.*

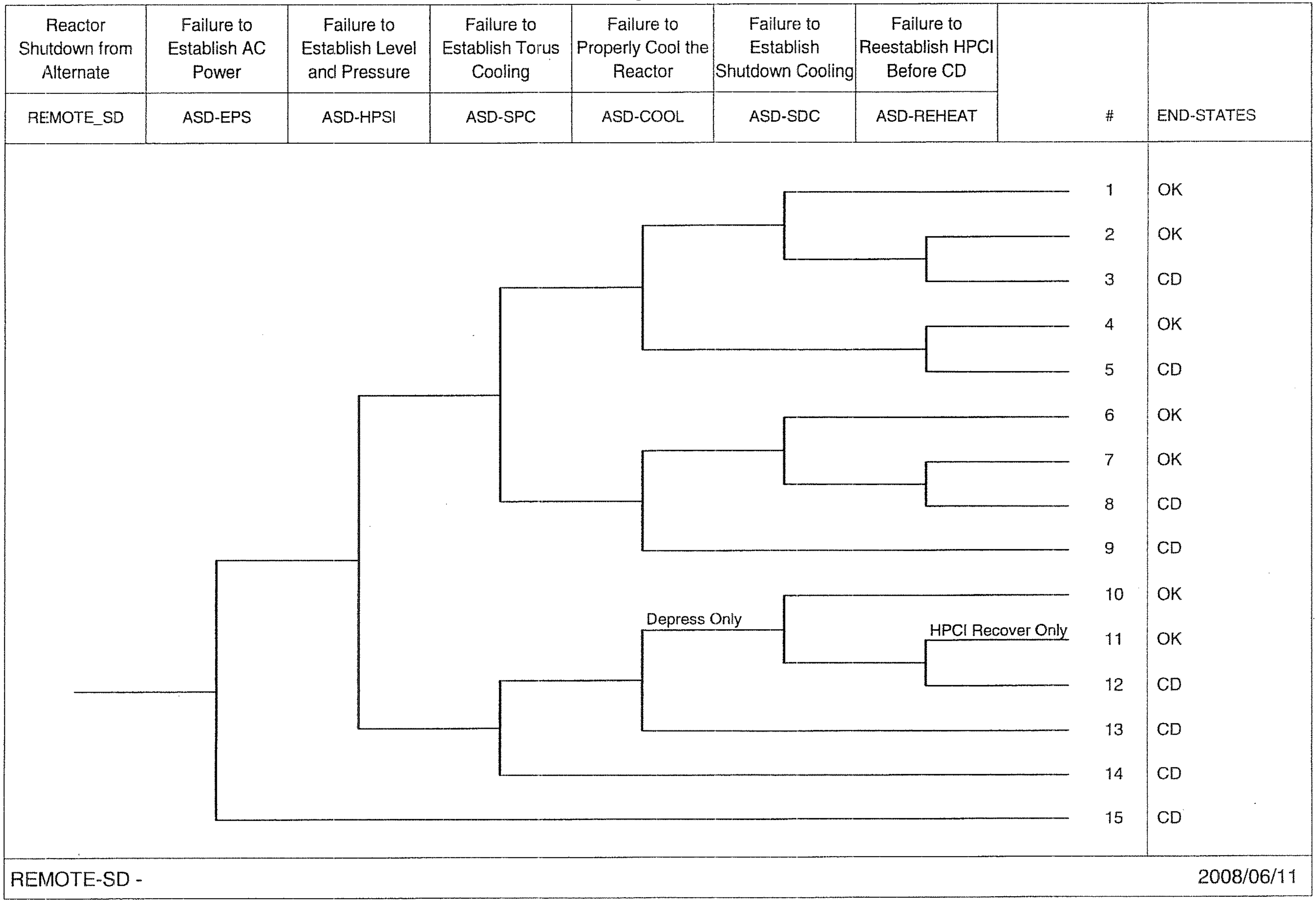
In the licensee's best case evaluation of the performance deficiency, the suppression pool would remain above 150°F for 10.6 hours.

The licensee used a case-specific combined recovery in assessing the risk of this performance deficiency. Most of the recoveries discussed by the licensee would have been available with or without the performance deficiency. Therefore, these should be in the baseline model and portions of the sequences subtracted from the case evaluation. This is the approach used by the analyst in the linked event trees model.The licensee stated during the regulatory conference that credit should be given for diesel-driven fire water pump injection. This is one of the licensee's alternative strategies. However, the inspectors determined, and the licensee concurred, that this alternative method of injection requires that Valve RHR-MO-25B be open. Therefore, no credit was given for this alternative strategy.

Conclusions:

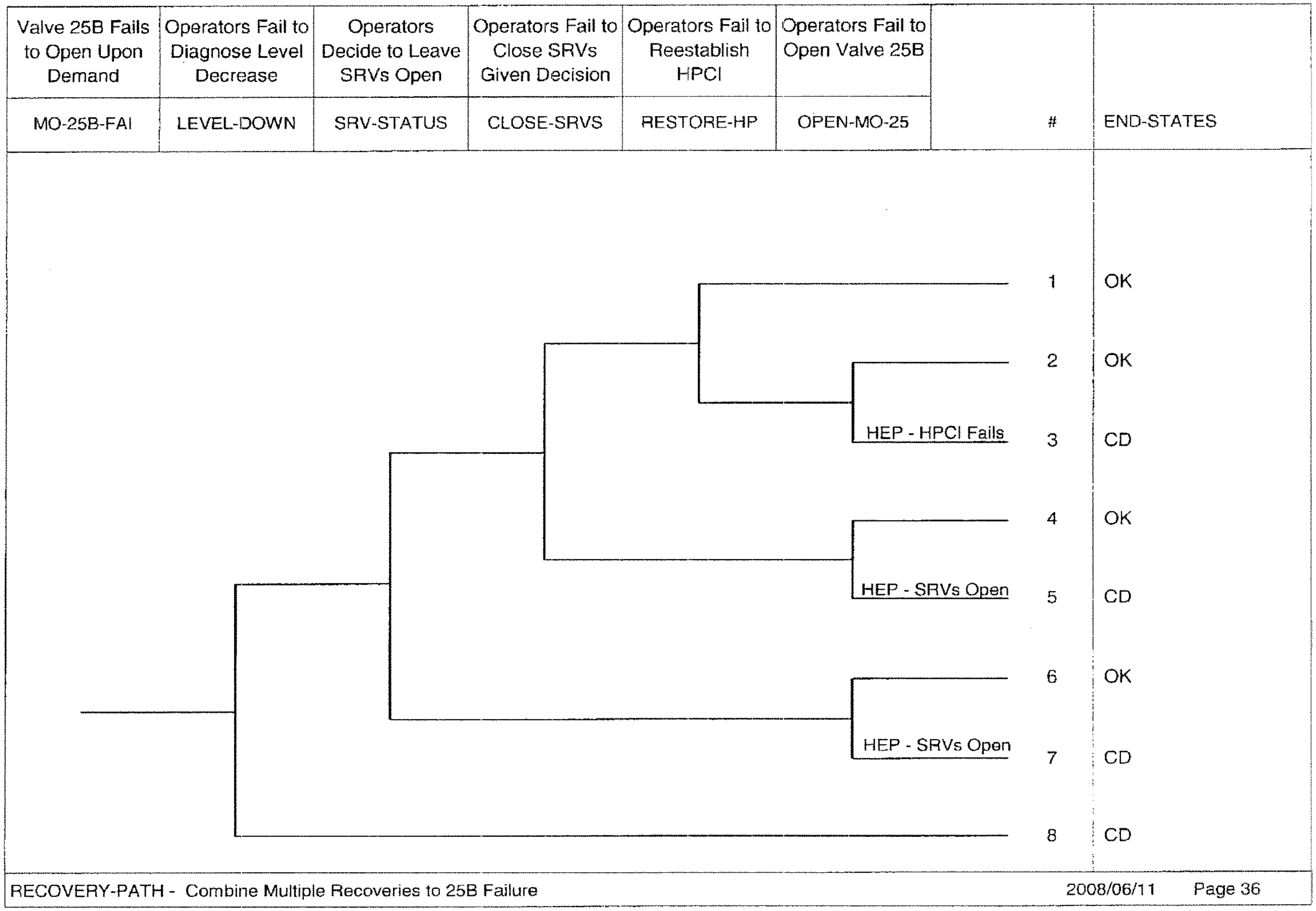
The analyst concluded that the performance deficiency was of low to moderate significance (White). As documented in Table 1, for a period of exposure of 1 year, the analyst determined a best estimate ΔCDF for fire scenarios that did not require evacuation of the main control room of less than 1.0E-7/yr. using both quantitative and qualitative techniques. Additionally, using the linked event tree model described in Assumption 4 for a period of exposure of 1 year, the analyst calculated the ΔCDF to be 2.0E-6/yr. for postulated fires leading to the abandonment of the main control room. This resulted in a total best estimate ΔCDF of 2.0E-6/yr.

Figure 1



Reactor Shutdown from Alternate
Failure to Establish AC Power
Failure to Establish Level and Pressure
Failure to Establish Torus Cooling
Failure to Properly Cool the Reactor
Failure to Establish Shutdown Cooling
Failure to Reestablish HPCI Before CD
REMOTE_SD
ASD-EPS
ASD-HPSI
ASD-SPC
ASD-COOL
ASD-SDC
ASD-REHEAT
#
END-STATES
1 OK
2 OK
3 CD
4 OK
5 CD
6 OK
7 OK
8 CD
9 CD
10 OK
11 OK
12 CD
13 CD
14 CD
15 CD
Depress Only
HPCI Recover Only
REMOTE-SD -
2008/06/11

Figure
2



Valve 25B Fails to Open Upon Demand
Operators Fail to Diagnose Level Decrease
Operators Decide to Leave SRVs Open
Operators Fail to Close SRVs Given Decision
Operators Fail to Reestablish HPCI
Operators Fail to Open Valve 25B
MO-25B-FAI
LEVEL-DOWN
SRV-STATUS
CLOSE-SRVS
RESTORE-HP
OPEN-MO-25
#
END-STATES
HEP - HPCI Fails
HEP - SRVs Open
HEP - SRVs Open
1 OK
2 OK
3 CD
4 OK
5 CD
6 OK
7 CD
8 CD
RECOVERY-PATH - Combine Multiple Recoveries to 25B Failure
2008/06/11
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