UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION IV
612 EAST LAMAR BLVD, SUITE 400
ARLINGTON, TEXAS 76011-4125

June 10, 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: FINAL SIGNIFICANCE DETERMINATION OF WHITE FINDING AND NOTICE OF VIOLATION, NRC INSPECTION REPORT 05000298/2011009, COOPER NUCLEAR STATION

Dear Mr. O'Grady:

This letter provides you the final results of our significance determination of the preliminary White finding identified in our letter dated March 17, 2011, which included the Nuclear Regulatory Commission's (NRC) Inspection Report 05000298/2010006. The finding involves the failure to verify that procedure steps to safely shut down the plant in the event of a fire would actually reposition three motor-operated valves to the required positions, while addressing a previous finding that involved the same procedure steps.

At your request, a regulatory conference was held on April 27, 2011, to further discuss your views on this issue. A copy of the handout you provided is attached. During the meeting, your staff described your assessment of the significance of the finding and the corrective actions taken to resolve it, including the root-cause evaluation of the finding.

During the Regulatory Conference, your staff presented specific differences between Nebraska Public Power District's risk assessment and the NRC's preliminary risk assessment of the finding. You stated the following positions: (1) the nominal time required to diagnose the failure of valve MOV-RHR-25B to stroke open should be 5 minutes; (2) the failure of high pressure coolant injection (HPCI) after 4 hours should not be considered, since more time for operator recovery would be available if HPCI had been successful for that long; (3) HPCI failure because of loss of room cooling or loss of ac power sources should be removed from consideration, since these would not fail HPCI in the first 4 hours; and (4) control rod drive pumps and feed and condensate pumps might be available to provide reactor makeup.

Following extensive review of the information you provided at the regulatory conference and detailed analysis of the probabilistic risk of the finding, the NRC concluded that the finding is appropriately characterized as White, an issue with low to moderate increased importance to safety.

The final significance determination, described in Enclosure 2, was based on the significance determination process Phase 3 analysis performed by the NRC staff using multiple risk tools including a linked event tree model of the Cooper Nuclear Station's remote shutdown capabilities developed by NRC analysts in 2008, to evaluate a White finding involving procedural problems that affected 10 valves, including the 3 valves addressed by this finding. The previous evaluation was documented in Notice of Violation EA 07-204, NRC Inspection Report 05000298/2008008, dated June 13, 2008. Corrective actions for the previous White finding included revisions to the procedure used for fires requiring control room evacuation. However, as the cause of the previous failure was not adequately determined, the corrective actions failed to completely correct the condition adverse to quality. The procedure improvements resulted in decreasing the non-recovery probability by a factor of 78 for the current finding where HPCI was successful.

Regarding the four points that your staff presented at the regulatory conference related to the probabilistic risk of the finding, the NRC staff considered this information in determining the final significance. In regards to the 5-minute diagnosis time, the NRC concluded that the points you raised, including the video demonstration, did not provide sufficient evidence to support a 5-minute nominal diagnosis time when consideration was given to the other factors that contribute to successful diagnosis, including procedure quality and operator training. Therefore, we have decided that a 15-minute nominal time to diagnose the failure of valve MOV-RHR-25B remains the most appropriate value to use.

The NRC evaluated your perspectives regarding the risk impact of a HPCI failure and agreed that scenarios where HPCI failed following 4 hours of run time would provide longer times for recovery. However, the NRC disagreed that these scenarios should be discounted as negligible. Therefore, the NRC divided the core damage sequences into two independent bins. For the first 4 hours, a loss of room cooling or a loss of ac power would not cause HPCI failure; the human error probability (HEP) used in the preliminary analysis (0.72) was retained as the appropriate value. For HPCI failures between 4 and 24 hours, the staff changed the performance shaping factor for available time by assigning "nominal time" for both the diagnosis and action events. All other performance shaping factors were left unchanged. This resulted in an HEP of 0.17.

You also stated that control rod drive pumps and feed and condensate pumps might be available to provide reactor makeup. Our reviews confirmed that these sources would not be available in the evaluated scenarios, so we did not credit them.

In summary, after considering the information developed during this inspection and the additional information your staff provided at the regulatory conference, the NRC has concluded that the inspection finding is appropriately characterized as White, an issue with low to moderate increased importance to safety, which may require additional NRC inspections.

You have 30 days from the date of this letter to appeal the staff's determination of significance for the identified White finding. Such appeals will be considered to have merit only if they meet the criteria given in the NRC Inspection Manual Chapter 0609, "Significance Determination Process," Attachment 2, "Process for Appealing NRC Characterization of Inspection Findings (Significance Determination Process Appeal Process)." An appeal must be sent in writing to the Regional Administrator, Region IV, Nuclear Regulatory Commission, 612 East Lamar Boulevard, Suite 400, Arlington, Texas 76011-4125.

The NRC has also determined that, together, the failure to correct the previous inadequate procedure problem and having inadequate emergency procedures involves a violation of NRC requirements as cited in the enclosed Notice of Violation (Notice). The circumstances surrounding the violation are described in detail in NRC Inspection Report 05000298/2010006. This violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," and 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," involved a repeat failure to ensure that some steps contained in emergency procedures at the Cooper Nuclear Station would work as written.

Specifically, steps in Emergency Procedures 5.4 POST-FIRE, "Post Fire Operational Information," and 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. As a consequence of this violation, these quality-related procedures would have challenged the operator's ability to bring the plant to a safe shutdown condition in the event of certain fires. In accordance with the NRC Enforcement Policy, the Notice is considered escalated enforcement action because it is associated with a White finding.

You are required to respond to this letter and should follow the instructions specified in the enclosed Notice when preparing your response. If you have additional information that you believe the NRC should consider, you may provide it in your response to the Notice. The NRC review of your response to the Notice will also determine whether further enforcement action is necessary to ensure compliance with regulatory requirements.

Because plant performance for this issue has been determined to be beyond the licensee response band, we will use the NRC's Action Matrix to determine the most appropriate NRC response for this event. We will notify you, by separate correspondence, of that determination.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosures, and your response 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,

Elmo E. Collins

Elmo E. Collins
Regional Administrator

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

Enclosures:
1. Notice of Violation
2. Final Significance Determination
3. Summary of Findings
4. Supplemental Information
5. License Regulatory Conference Slides

cc w/enclosures:
Distribution via ListServe for CNS

NOTICE OF VIOLATION

Nebraska Public Power District
Cooper Nuclear Station

Docket No. 50-298
License No. DPR-46
EA-11-024

During an NRC inspection completed on March 14, 2011, a violation of NRC requirements was identified. In accordance with the NRC Enforcement Policy, the violation is listed below:

> Title 10 of the Code of Federal Regulations, Part 50, Appendix B, Criterion XVI, "Corrective Actions," requires, in part, that 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.
>
> 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.
>
> 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.
>
> Contrary to the above, between July 1997 and November 2010, the licensee failed to establish measures to assure a condition adverse to quality was corrected and ensure the activities affecting quality were prescribed by documented procedures appropriate to the circumstances. Specifically, Violation 05000298/2008008-1, dated June 13, 2008, identified a condition adverse to quality in that Emergency Procedures 5.4 POST-FIRE and 5.4 FIRE-SD would not work as written. While correcting that violation, the licensee failed to perform sufficient evaluation of the circuits to identify and correct a problem with three motor-operated valves needed to establish core cooling, RHR-MOV-25A, RHR-MOV-25B and RHR-MOV-53A. Failure to correct the condition adverse to quality resulted in inadequate procedures in that they contained steps that were inappropriate to the circumstances because they would not work as written to reposition the three motor-operated valves.

This violation is associated with a White significance determination process finding.

Pursuant to the provisions of 10 CFR 2.201, Nebraska Public Power District is hereby required to submit a written statement or explanation to the U.S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001 with a copy to the Regional Administrator, Region IV, 612 East Lamar Blvd., Arlington, TX 76011-4125, and a copy to the NRC Resident Inspector at the Cooper Nuclear Station within 30 days of the date of the letter transmitting this Notice of Violation (Notice). This reply should be clearly marked as a "Reply to a Notice of Violation: EA-11-024," and should include: (1) the reason for the violation, or, if contested, the basis for disputing the violation, (2) the corrective steps that have been taken and the results achieved, (3) the corrective steps that will be taken, and (4) the date when full compliance will be achieved. Your response may reference or include previous docketed correspondence, if the correspondence adequately addresses the required response. If an adequate reply is not received within the time specified in this Notice, an order or a Demand for Information may be issued as to why the license should not be modified, suspended, or revoked, or why such other action as may be proper should not be taken. Where good cause is shown, consideration will be given to extending the response time.

If you contest this enforcement action, you should also provide a copy of your response, with the basis for your denial, to the Director, Office of Enforcement, United States Nuclear Regulatory Commission, Washington, DC 20555-0001.

Because your response 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's website at www.nrc.gov/reading-rm/adams.html, to the extent possible, it should not include any personal privacy, proprietary, or safeguards information so that it can be made available to the public without redaction. If personal privacy or proprietary information is necessary to provide an acceptable response, then please provide a bracketed copy of your response that identifies the information that should be protected and a redacted copy of your response that deletes such information. If you request withholding of such material, you must specifically identify the portions of your response that you seek to have withheld and provide in detail the bases for your claim of withholding (e.g., explain why the disclosure of information will create an unwarranted invasion of personal privacy or provide the information required by 10 CFR 2.390(b) to support a request for withholding confidential commercial or financial information). If safeguards information is necessary to provide an acceptable response, please provide the level of protection described in 10 CFR 73.21.

Dated this 10th day of June 2011

FINAL SIGNIFICANCE DETERMINATION

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 stated 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, Attachment 3, Appendix F, 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 10 CFR Part 50, 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 5.4POST-FIRE and 5.4FIRE-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 5.4POST-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 5.4FIRE-S/D. This is the credited train and the only procedural means for initiating alternate shutdown cooling during the recovery actions. Changes were made to Procedure 5.4FIRE-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 5.4FIRE-S/D. These changes in Attachment 1 of the procedure directed the operator at the remote shutdown panel to close safety relief valves (SRVs) if residual heat removal (RHR) injection was not observed to be successful and stabilize conditions using high pressure coolant injection (HPCI). 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 valve 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 preliminary delta-CDF to be 2.0E-6/yr for postulated fires leading to the abandonment of the main control room.

At the licensee's request, a regulatory conference was held on April 27, 2011. During the meeting, the licensee described their assessment of the significance of the finding and the corrective actions taken to resolve it, including the root cause evaluation of the finding.

The licensee disagreed with the NRC analysts' assumptions concerning the time to diagnose the failure of valve MOV-RHR-25B to stroke open and the analysis of HPCI failures. The licensee presented information and a video demonstration supporting their position that the NRC human error probability (HEP) value of 0.72 to open valve MOV-RHR-25B manually following an early HPCI failure was too high. This was based on a disagreement in selecting the nominal time value necessary to diagnose the failure of valve MOV-RHR-25B to stroke open.

The licensee stated that the nominal time for diagnosis should be 5 minutes, rather than 15 minutes as documented in the Phase 3 analysis. Both the licensee and the NRC agreed that 5 minutes should be reserved for the action to locally open the valve, leaving the time available for diagnosis of 10 minutes. The NRC asked the licensee to describe the uncertainty associated with the thermal-hydraulic calculation used to estimate the total time available to diagnose and implement recovery actions (15 minutes). The licensee did not have an uncertainty analysis, but believed it to be an appropriate best-estimate value. The staff concluded that the licensee did not provide sufficient evidence that there was no uncertainty in the total time available to diagnose and implement recovery actions.

The NRC also determined that the licensee did not provide sufficient evidence to support a 5-minute average diagnosis time. The NRC determined that the operator in the video demonstration was not following the procedure as written and was performing steps before they were called for in the procedure, which artificially reduced the time to reach a diagnosis. The NRC determined that neither the procedure nor biennial operator training covered the situation being evaluated, making it unreasonable to consider the single demonstration to represent average performance. The procedure was not symptom-based or diagnostic, and would not lead operators to the correct diagnosis or required recovery action. Each of these elements must be considered to assess average time to diagnose.

Additionally, the operator did not appear to consider other possible causes for the unexpected flow indications, such as: the throttle position of valve MOV-RHR-34B; the return valve to the suppression pool; or the need to further manipulate the safety relief valves to adjust pressure in the reactor pressure vessel; retrieve prints to analyze other potential failures; or, as stated in the procedure, the operator might return to an earlier step and repeat the steps in sequence. This potentially could result in another attempt by the reactor building operator to open valve MOV-RHR-25B from the motor starter. Also, the operator might be concerned about recovering HPCI and focus on that goal instead of diagnosing the failure to start injection flow. In consideration of these points, the NRC concluded that a nominal diagnosis time of 15 minutes was reasonable.

The licensee also stated that control rod drive pumps and feed and condensate pumps might be available to provide reactor makeup. Our reviews confirmed that these sources would not be available in the evaluated scenarios, so we did not credit them.

The licensee also stated that there were aspects of HPCI failures that would not result in a restricted time frame (previously determined by analysis to be 15 minutes) for diagnosing and recovering valve MOV-RHR-25B. One example given was that the fail-to-run event considers the entire failure probability for 24 hours of operation, but if HPCI runs for 4 hours or greater, the time available to recover valve MOV-RHR-25B would be greater than 15 minutes and would offer a better probability of success. Other examples where the postulated HPCI failure would be beyond 4 hours included losses of room cooling and ac power sources, which were not specifically applicable to the scenario of concern.

The NRC agreed that scenarios where HPCI failed following 4 hours of run time would provide longer times for recovery. However, the NRC disagreed that these scenarios should be discounted as negligible. Therefore, the NRC divided the core damage sequences into two independent bins:

1. Postulated failure of HPCI was within the first 4 hours

   Bin 1 sequences included failure of the HPCI system to start and/or realign and failure of the HPCI pump to run for the first 4 hours. Sequences involving failures of HPCI caused by a loss of room cooling or a loss of ac power were not included in Bin 1 because the calculated failure time for the HPCI system in these cases was beyond 4 hours. For these sequences, the staff concluded that the HEP used in the preliminary analysis (0.72) was appropriate.

2. Postulated failure of HPCI was from 4 to 24 hours.

   Bin 2 sequences included any failure of the HPCI system to run with a postulated failure beyond 4 hours. These sequences included the HPCI pump failure-to-run basic event modified to account for pump failures from 4 to 24 hours, HPCI failure as a result of loss of room cooling, and HPCI failure following battery depletion caused by a loss of ac power. For these sequences, the staff changed the performance shaping factor for available time by assigning "nominal time" for both the diagnosis and action events. All other performance shaping factors were left unchanged.

The result is shown in the following table:

	Diagnosis (nominal =1.0E-2)	Action (nominal = 1.0E-3)
Available Time	Nominal	Nominal
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	20	5.5
HEP	0.168	0.005
Total HEP	0.17	

This resulted in the following assignments of HEP values for manually opening MOV-RHR-25B:

	Non-Recovery Value
HPCI Success	1.01E-3
HPCI Failure Cable Spreading Room	1.1E-2
Early HPCI Failure (0-4 Hours) All Other ASD Areas	0.72
Late HPCI Failure (4-24 Hours) All Other ASD Areas	0.17

The analyst hand-calculated the resulting delta-CDF by placing the above HEP values into the appropriate cutsets. The result was a delta-CDF of 1.2E-6/yr, therefore, the NRC has concluded that the inspection finding is appropriately characterized as White, an issue with low to moderate increased importance to safety

SUMMARY OF FINDINGS

IR 05000298/2011009; March 14, 2011 – June 10, 2011; Nebraska Public Power District; Cooper Nuclear Station: Triennial Fire Protection Team Inspection.

This report covers an inspection follow-up and significance determination effort by region-based inspectors and senior risk analysts. One finding was identified with an associated violation, which was determined to have low-to-moderate safety significance (White). 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. <u>NRC-Identified and Self-Revealing Findings</u>

Cornerstone: Mitigating Systems

<u>White</u>. A violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action," and Criterion V, "Instructions, Procedures, and Drawings," was identified for failure to ensure that some steps contained in Emergency Procedures at the 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 Procedures 5.4 POST-FIRE, "Post-Fire Operational Information," and 5.4 FIRE-S/D, "Fire Induced Shutdown From Outside Control Room," intended to reposition three 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 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 correct 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 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.

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee Personnel

D. Buman, Director of Engineering
S. Nelson, Risk Supervisor
B. O'Grady, Vice President-Nuclear and Chief Nuclear Officer
J. Olberding, Nuclear Support
R. Penfield, Operations Manager
A. Zaremba, Director, Nuclear Safety Assurance

NRC personnel

E. Collins, Regional Administrator, Region IV
J. Josey, Senior Resident Inspector
R. Kellar, Senior Enforcement Specialist, RIV
D. Loveless, Senior Reactor Analyst, RIV/DRS
J. Mateychick, Senior Reactor Inspector, RIV/DRS
N. O'Keefe, Chief, Engineering Branch 2, RIV/DRS
T. Pruett, Deputy Director, RIV/DRP
M. Runyan, Senior Reactor Analyst, RIV/DRS
A. Vegel, Director, RIV/DRS

LIST OF ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000298/2011009-01	VIO	Inadequate Post-Fire Safe Shutdown Procedures (EA-11-024)

Closed

05000298/2010006-01	AV	Inadequate Post-Fire Safe Shutdown Procedures (Section 1R05.01)

Enclosure 5

Licensee Regulatory Conference Slides

Regulatory Conference

Triennial Fire Protection

IR 2010-06

Cooper Nuclear Station

April 27, 2011

Introductions and Opening Comments

Brian O'Grady

Vice President – Nuclear and Chief Nuclear Officer

Agenda

- Root Cause
- Corrective Actions
- Risk Significance
- Closing Remarks

Previous Corrective Actions

Rod Penfield
Operations Manager

Problem Statement

Failure to ensure some steps contained in emergency procedures would work as written and the condition was not promptly identified and corrected.

Previous Violation

- Root Cause Determination was Adequate
  - Safe Shutdown Analysis Report not Validated
  - Field Verification of Procedure not Adequate
- Corrective Actions Narrowly Focused and Inadequate
  - Future Looking Only
  - Procedure Validations did not Ensure Intended Results

Corrective Actions Going Forward

- Physical Validation Required
- Procedures Revised to Eliminate Contactor Operation
- Mock Up Built to Simulate all Types of Contactors
- Modification Funded
- Additional Guidance for Appendix R Design Review

Modifications

- Benchmarked Other Sites
  - Cooper is Outlier
- We Commit to Modify the Plant by Next Outage
  - New Valve Control
    - Install New Isolation Switches
    - Install New Remote Control Switches
    - Install New Remote Indication
- Eliminates Operators Pushing Contactors
- Will be Included in Surveillance Testing Program

Risk Significance

Steve Nelson
Risk Management Supervisor

Risk of Fires Leading to Control Room Abandonment

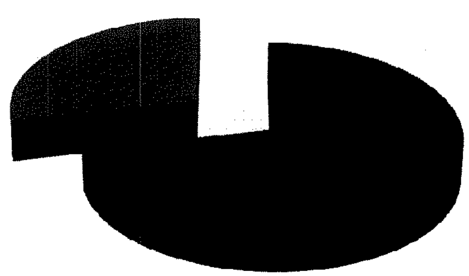
Improved Procedure Guidance
2007 vs. 2010 Risk Comparison

With HPCI operating as designed	5.20E-06	< 1.0E-07	99.6%
With HPCI Failure	2.10E-06	1.98E-06	5.7%
Total	7.30E-06	2.00E-06	72.6%

HPCI Contribution to Risk of Performance Deficiency

RHR-MOV-MO25B Delta CDF

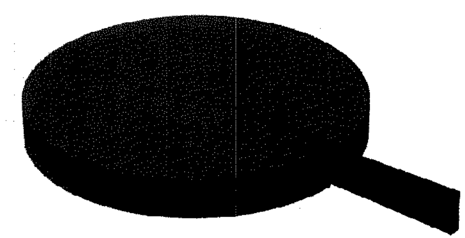
2007 HPCI Success Risk Contribution



■ HPCI Success

■ HPCI Failure

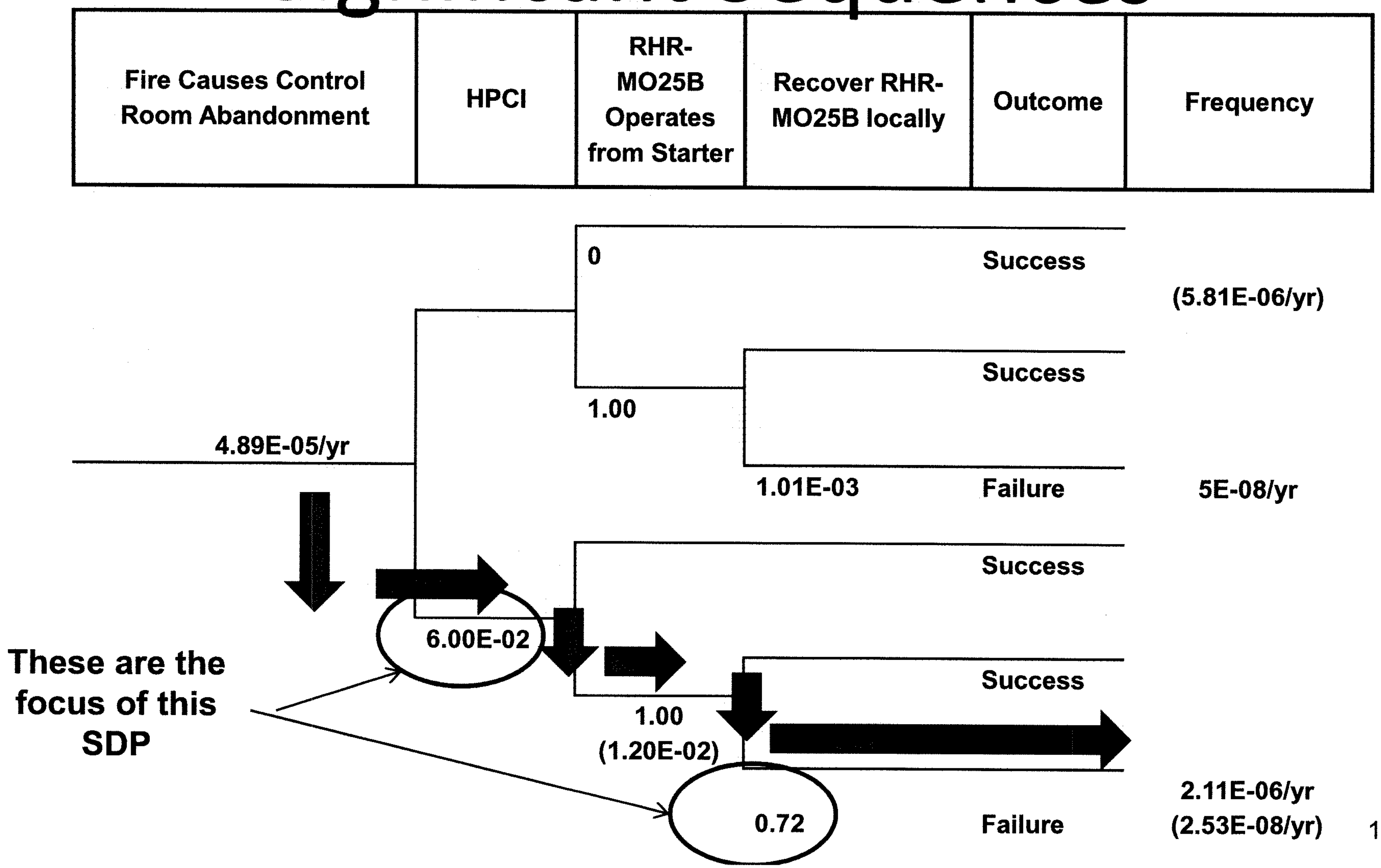
2010 HPCI Success Risk Contribution



■ HPCI Success

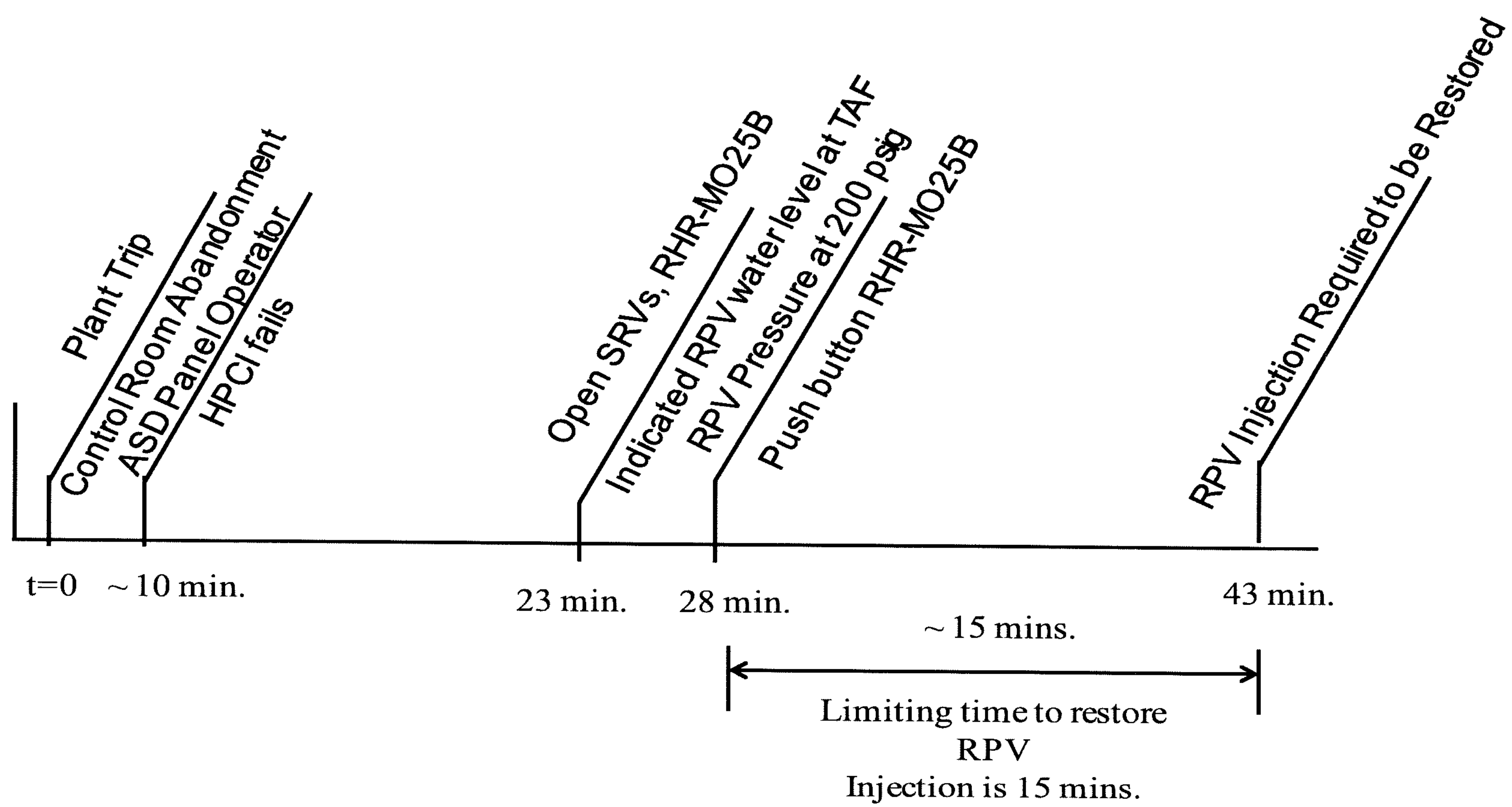
■ HPCI Failure

Understanding the Risk Significant Sequences



Time Line Comparisons

Time Line for Fire Causing Control Room Abandonment without credited HSD path available



HPCI Failure Probability

	Cooper	NRC
HPCI Failure Probability (All failure modes - 24 hrs)	5.69 E-02/yr	6.00E-02/yr
HPCI Failure Probability (early failures – 4 hrs)	3.50 E-02/yr	Not evaluated
Risk Significance (% reduction of NRC delta CDF)	1.23E-06/yr (38.5% reduction)	2.00E-06/yr (Not Evaluated)

Human Error & SPAR – H Model

Total nominal HEP =

Sum of HEP (diagnosis) + HEP (action)

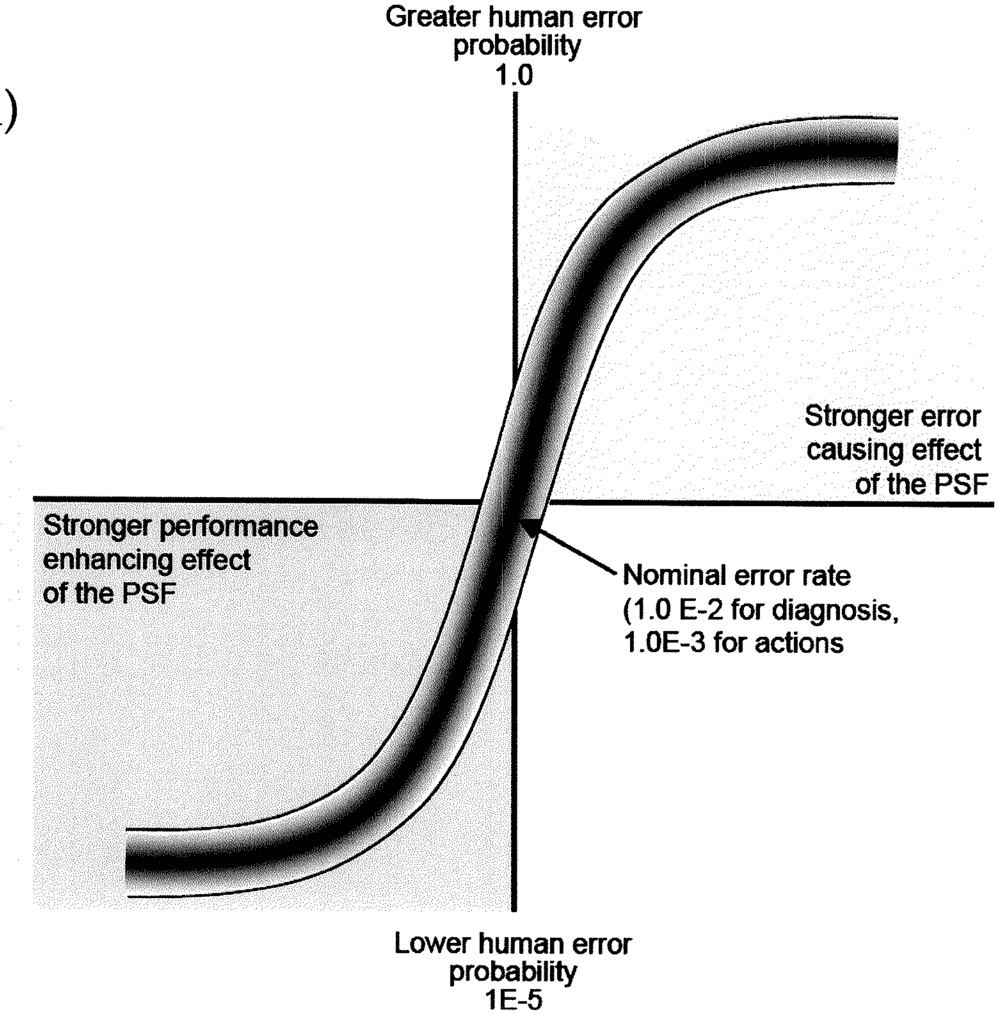
Nominal values:

HEP (diagnosis) = 1.0E-02

HEP (task) = 1.0E-03

HEP (total) = 1.1E-02

- PSF impact is evaluated for both the diagnosis and task portions.
- The major difference in the risk evaluations involve the diagnosis portion of the HEP.



Human Error Probability - Diagnosis

Performance Shaping Factor	NRC Value	Cooper Value	Comparison: NRC to Cooper
Available Time	10 – Barely Adequate (2/3 nominal)	1 - Nominal	10 x
Stress	2 - High	2 – High	None
Complexity	2 – Moderate	1 – Nominal	2 x
Experience/Training	1 - Nominal	0.5 - High	2 x
Procedures	5 – Poor	0.5 – Diagnostic/symptom oriented	10 x
Ergonomics	1 - Nominal	1 – Nominal	None
Total PSF Product	200 = 10 x 2 x 2 x 5	0.5 = 2 x 0.5 x 0.5	400 x

Human Error Probability - Action

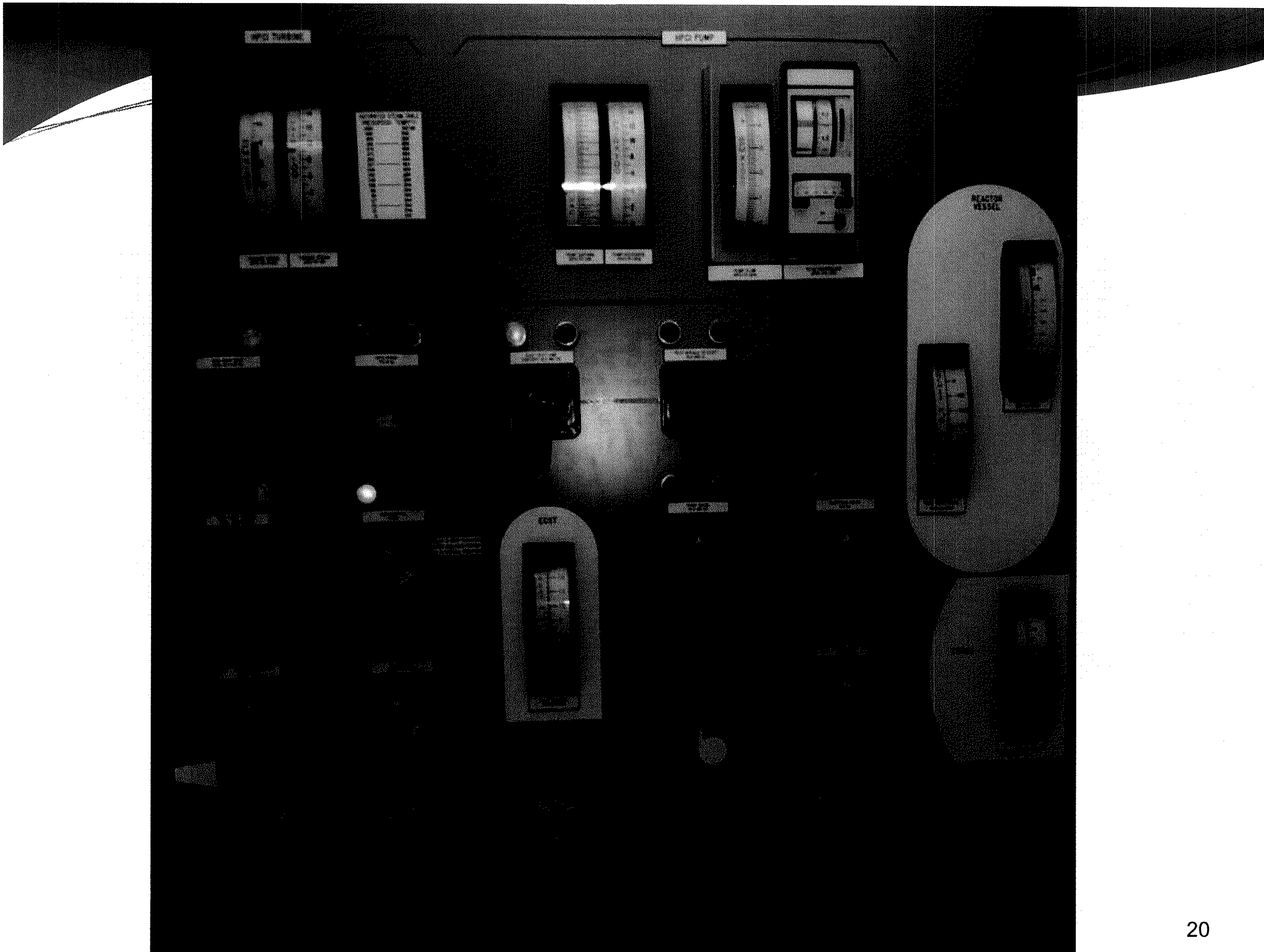
Performance Shaping Factor	NRC Value	Cooper Value	Comparison: NRC to Cooper
Available Time	10 – Time Required	1 - Nominal	10 x
Stress	2 - High	2 – High	None
Complexity	1 - Nominal	1 - Nominal	None
Experience/Training	0.5 - High	0.5 - High	None
Procedures	1 - Nominal	1 - Nominal	None
Ergonomics	5.5 - 50% Poor, 50% Nominal	1 0 – Poor	0.55 x
Total PSF Product	55 = 10 x 2 x 0.5 x 5.5	10 = 2 x 0.5 x 10	5.5 x

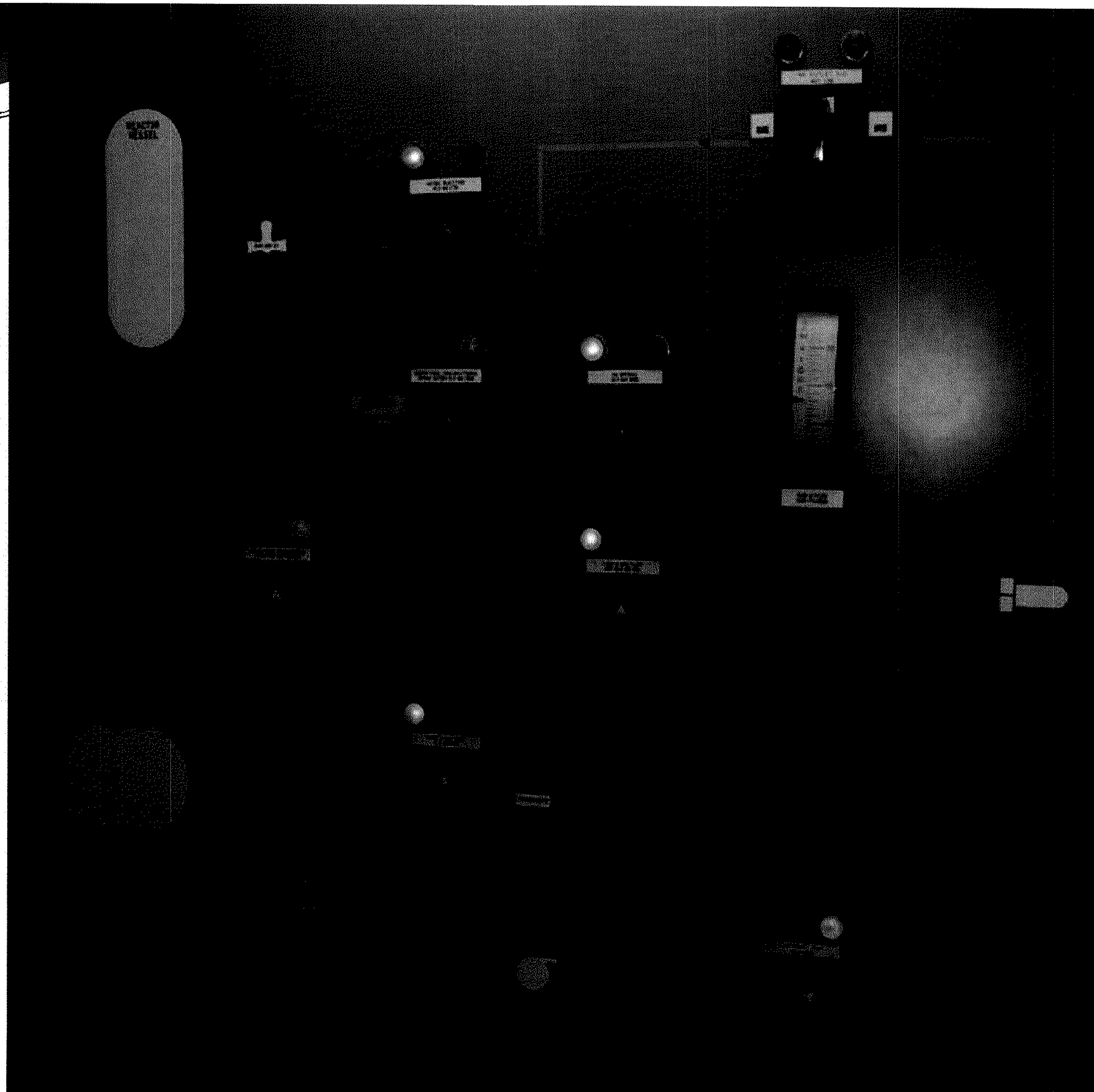
Diagnosis – Available Time

- NUREG / CR 6883 Definitions
  - Barely Adequate – *"2/3 the average time to diagnose the problem is available"*
  - Nominal – *"On average there is sufficient time to diagnose the problem"*
- Diagnosis HEP (nominal time) = 0.168 vs. 0.67
- Total HEP = 0.168 + 0.05 = 0.218
- Risk Significance ~ 6E-07/year vs. 2E-06/year

Risk Assessment Comparisons

- Agreements
  - When HPCI is successful, risk significantly reduced due to procedure improvements
  - HPCI failure bounding timeline is worst case
  - Average action time approximately 5 minutes to open valve
- Differences
  - HPCI failure modes greater than 4 hours should not apply
  - Operator recognition of RHR injection failure is straight forward
  - Average diagnosis time should be on the order of 5 minutes not 15 minutes





Diagnosis

- Adequate Indications at ASD Panel
  - RHR Flow Indication
  - RPV Water Level
  - Valve Indications
- Reactor Level Indication
  - Shrink Due to Cold Water will not Cause Misdiagnosis
  - Procedure Guidance for SRV Control will not Mask Level Indication
- Operator Training

Risk Significance Conclusions

- Change in Core Damage Frequency for Early HPCI Failures is less than 3.36E-07/yr, when using Estimates of:
  - HPCI Failure Probability of 3.5%
  - Average Diagnosis Time is Less than 5 Minutes

Closing Remarks

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