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

MAY 28 2008

Stewart B. Minahan, Vice
President-Nuclear and CNO
Nebraska Public Power District
P.O. Box 98
Brownville, NE 68321

SUBJECT: MEETING SUMMARY FOR REGULATORY CONFERENCE WITH NEBRASKA PUBLIC POWER DISTRICT

Dear Mr. Minahan,

This refers to the Regulatory Conference public meeting conducted at the Nuclear Regulatory Commission, Region IV office on May 13, 2008. This conference related to the discussion of the significance of an inspection finding associated with two procedures that were not adequate to be used by operators to bring the plant to a safe shutdown condition in the event of certain fire scenarios. The finding was documented in inspection report 2008007. The meeting attendance list and a copy of the slides presented by you and your staff during the meeting are included as Enclosures 1 and 2, respectively.

In accordance with Section 2.390 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and its enclosure will be available electronically for public inspection in the NRC's Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS). ADAMS is accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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

Sincerely,

Gregory Pick for

Linda J. Smith, Chief
Plant Engineering Branch
Division of Reactor Safety

Docket: 50-298
License: DPR-46

Enclosures:
1. Meeting Attendance List
2. NPPD Staff Presentation

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Ronald D. Asche, President
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Enclosure 1
Meeting Attendance List

Regulatory Conference - Category 1 Public Meeting	
LICENSEE/FACILITY	Nebraska Public Power District - Cooper Nuclear Station
DATE/TIME	May 13, 2008 08:00-11:00
CONFERENCE LOCATION	Region IV Office, Training & Conference Room (TCR)
EA NUMBER	07-204

~~NRC REPRESENTATIVES~~ BC 5/13/08

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Neil O'Keefe	NRC	Sr Rx Inspector
Roy Caniano	NRC	D:DRS
Troy Pruett	NRC	DD:DRS
Elmo Collins	NRC	RA:RIV
Mark Cunningham	NRC	D:NRR/DRA
Alex Klein	NRC	BC:APOB
DAVID LOVELESS	NRC	SR. REACTOR ANALYST
JEFF CIRCLE	NRC	SR. REL./RISK ANALYST
Michael Vasquez	NRC	Sr. Enf. Specialist
DOUG STARKEY	NRC	SR. ENFORCEMENT SPEC
Rod Penfield	NPPD	ASSISTANT OPS. MGR.
KENT SUTTON	NPPD	RISK MGMT SUPR
John McCann	ENTERGY	Director - Licensing
SB Minahan	Entergy / NPPD	SITE VP
Paul V Fleming	NPPD	Director Nuclear Safety Assurance
GARY KLINE	NPPD	DIRECTOR / ENGINEERING
JODIE [illegible]	NPPD	LICENSING SPECIALIST

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NAME (PLEASE PRINT)	ORGANIZATION	TITLE
Eric Ruesch	NRC/DRS	Reactor Inspector
Karla Fuller	NRC/ACES	Regional Counsel/TL ACES
Steven Alferink	NRC/DRS	Reactor Inspector
Hasan Abuseini	NRC/DRS	Reactor Inspector
Nnaerika Okonkwo	NRC/DRS	Reactor Inspector
Eduardo Uribe	NRC/DRS	Reactor Inspector
Brian Correll	NRC/DRS/EB2	Reactor Inspector
Jeff Mitman	NRC/NRR	
Antonio Zoulis	NRC/NRR	
Robert Taylor	NRC/NRR	
Fred Lyon	NRC/NRR	
Paul Lain	NRC/NRR	
Harry Barrett	NRC/NRR	
Daniel Frumkin	NRC/NRR	
SeeMeng Wong	NRC/NRR	
Mary Ann Ashley	NRC/NRR	Enforcement Coordinator

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~~MEMBERS OF THE PUBLIC~~ BC 5/13/08

NAME (PLEASE PRINT)	ORGANIZATION	TITLE
LINDA JOY SMITH	USNRC	Chief. EB2
John M Mateychick	USNRC	Senior Reactor Inspector
Phil Qualls	NRC	Reactor Inspector
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James Blum	EPM inc.	Fire Protection Engineer
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Virgel T. Furr	NPPD	Risk Management Eng.
ROMAN M ESTRADA	NPPD	CORRECTIVE ACTION MANAGER
Thomas C. Poindexter	Morgan Lewis for NPPD	Counsel to NPPD
Kevin P. Billesbach	NPPD	QA MANAGER
David VanDerKamp	NPPD	Licensing Manager

Enclosure 2
NPPD Staff Presentation

PRE-DECISIONAL REGULATORY CONFERENCE

CONFERENCE WITH NEBRASKA PUBLIC POWER DISTRICT

MAY 13, 2008

NRC REGION IV, ARLINGTON, TEXAS

AGENDA

1.	Opening Remarks and Introductions	Roy Caniano, RIV, D:DRS
2.	Opening Remarks	Elmo Collins, Regional Administrator, RIV
3.	Apparent Violation and Summary of Interim Enforcement Discretion Policy Summary	Neil O'Keefe, RIV, EB2
4.	Significance Determination Results	David Loveless, Senior Reactor Analyst, RIV
5.	Licensee Presentation	Nebraska Public Power District Staff
6.	NRC Caucus	
7.	Resumption of Conference	
8.	Closing Remarks	Nebraska Public Power District Staff
9.	Closing Remarks	Roy Caniano, RIV, D:DRS NRC Staff
10.	Public Questions (NRC)	Roy Caniano, RIV, D:DRS

Regulatory Conference

Triennial Fire Protection
IR 2008-007

Cooper Nuclear Station

May 13, 2008

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Introductions and Opening Comments

Stewart Minahan
Vice President – Nuclear and
Chief Nuclear Officer

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Agenda

- Regulatory Discussion
  - Timeline
  - Key Issues
  - Enforcement Discretion
  - Process Issues
  - Summary
- Significance Determination Discussion
  - Performance Deficiency
  - New CNS Specific Information
  - Operator Response
  - Comparison
  - Summary
- Closing Remarks

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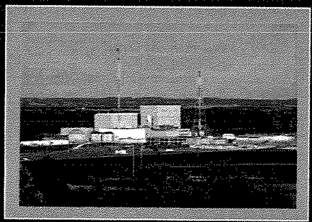
Regulatory Discussion

Paul V. Fleming

Director, Nuclear Safety Assurance



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Timeline

- June 15, 2007 – Onsite Debrief
  - 2 Findings with Enforcement Discretion
  - 1 Finding Needed Additional Review
- December 26, 2007 – Telephonic Exit
  - 2 Findings with Enforcement Discretion
  - 1 Finding Treated as URI to Complete Significance Evaluation
- February 1, 2008 – Inspection Report Issued
- March 18, 2008 – Telephonic Exit
  - Greater than Green, No Enforcement Discretion
- March 19, 2008 – Inspection Report Issued

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Key Issues

- Procedures Could Not Be Performed as Written
  - Manual Action Associated with Operation of Certain Valves
  - Inadequate Guidance
    - Contactor guidance in 1986
    - Fuse guidance in 1997
- NPPD
  - Procedures Have Been Corrected
- Application of Enforcement Discretion
- Application of Significance Determination

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Enforcement Discretion

- Enforcement Discretion Criteria
  - Four Criteria
    - Also must not be of high safety significance
  - All Must Be Satisfied
  - Three Criteria Not In Dispute
    - Has been determined to be less than high safety significance
  - Criterion 3 Recently Challenged

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Enforcement Discretion (cont.)

Criterion 3 States - It was not likely to have been identified by routine licensee efforts such as normal surveillance or quality assurance (QA) activities

- Important Aspects
  - Not Likely
  - Routine Effort
- Very Subjective Language
- “Routine” Processes Do Not Necessarily Equal “Routine” Effort

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Enforcement Discretion (cont.)

- NRC Determined
  - Weakness in "Routine" Process
    - Verification & Validation (V & V)
  - Inadequate Actions from 2004 Inspection
    - Contactor labeling for RHR-MO-25B
  - Missed Opportunities
    - QA Audit
    - Self-Assessment (Feasibility Study)
- Enforcement Discretion Denied

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Enforcement Discretion (cont.)

- CNS Evaluation - Verification & Validation Process
  - 1997 and 2004 V & V Appeared to be Comprehensive
  - CNS Uses Standard V & V Process
    - Benchmark
    - Documented in Root Cause
  - Performance of V & V is Not Routine Effort
    - Function of Change
    - Graded Approach
      - Complexity

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Enforcement Discretion (cont.)
CNS Evaluation – Inadequate Actions

- 2004 Inspection
  - Timing
    - High Pressure Coolant Injection (HPCI)
    - Main Steam Isolation Valves (MSIVs)
  - Method of Operation
    - Could Lead to Overthrust
    - Human Factors
- 2007 Inspection
  - Instructions Inconsistent with Field Configuration
    - Fuses
    - Contactors

Given Findings from 2004, it is Unlikely Conditions Identified in 2007 would have been Discovered using Routine Effort

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Enforcement Discretion (cont.)

- Corrective Action Program
  - "Routine" Process
  - When a Condition is Identified, Then it is Considered a Routine Effort to Generate a Condition Report
  - The Evaluation, Development and Implementation of Corrective Actions are Required Elements but Not Considered Routine Effort
    - Function of scope, complexity and resources

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Enforcement Discretion (cont.)

- CNS Evaluation - Missed Opportunities
  - Do Not Open Policy Limited Opportunity
  - 2007 Quality Assurance (QA) Audit
    - Effort involved ensuring manual actions can be achieved within required time by performing a plant walkdown and simulating actions
  - 2007 Self-Assessment (Feasibility Study)
    - Effort was a document review only and did not include any design verification or new walkdowns of the manual actions
  - Therefore it is Unlikely Conditions Identified in 2007 Would Have Been Discovered Using Routine Effort

Process Issues

- Consistency - Enforcement Discretion
  - Reviewed Other NRC Fire Protection Inspection Reports
    - 6 Violations/Non-compliances
    - 4 Plants/3 Regions
    - Transitioning to NFPA-805
    - 4th Quarter 2006 & 2007 Inspections
  - 3 of 6 were Procedure Issues

Process Issues (cont.)

- Consistency - Enforcement Discretion (cont.)
  - 3 Had Identified Missed Opportunities
  - Not Likely to Have Been Identified by Routine Licensee Effort
    - QA, Corrective Action Program and Self-Assessments
  - Potential Impacts Include Reactor Coolant System (RCS) Pressure Boundary, Auxiliary Feed Water Availability, Emergency Lighting
  - 6 of 6 Had Enforcement Discretion Granted
  - Generally Use Licensee Significance Evaluation

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Summary

- Enforcement Discretion Should Be Applied
  - Panels Not Routinely Opened
  - Routine Effort Would Not Likely Identify
- Based on Our Review of NRC Inspection Reports
  - Consistent Treatment is Not Apparent
    - Application of Enforcement Discretion
    - Application of Significance Determination

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Significance Determination Discussion
Performance Deficiency

Kent Sutton

Risk Management Supervisor

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Performance Deficiency Inadequate Guidance

- Two Procedures
  - 5.4POST-FIRE, Post-Fire Operational Information
    - Shutdown from the Control Room
    - 10 Valves
  - 5.4FIRE-S/D, Fire Induced Shutdown from Outside Control Room
    - 1 Valve (RHR-MO-25B)
- 10 Motor Operated Valves Affected
  - 5 Hot Shutdown (HSD), 5 Cold Shutdown (CSD)
    - 4 No Control Power Fuses
    - 6 Required Additional Contactors to be Pushed
- Valves Do Not Reposition Using Guidance as Written

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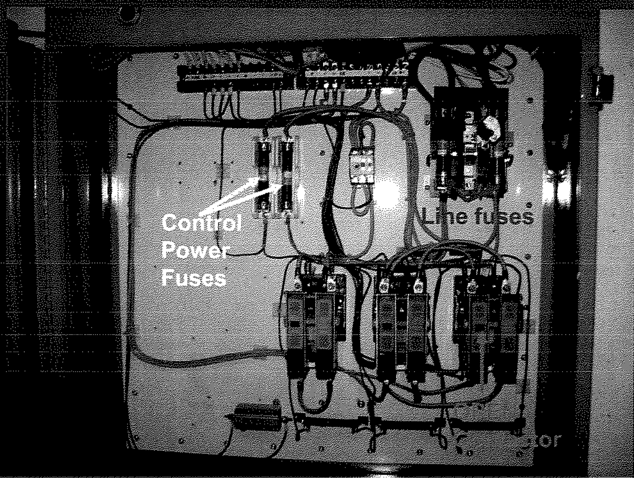
RHR-MO-25B



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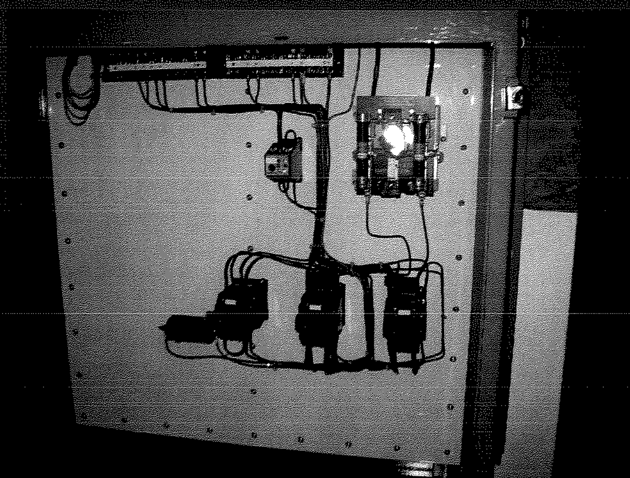
RWCU-MO-18



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Performance Deficiency Affected HSD Valves

- HPCI-MO-14, HPCI-MO-16
  - Procedure Did Not Identify Number of Contactors
  - Redundant Methods to Operate HPCI
- MS-MO-77, RHR-MO-921
  - No Effect on Safe Shutdown Train Success
    - Redundant Valves Exist on the Penetrations
- RWCU-MO-18
  - Closed Loop System

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Performance Deficiency Affected CSD Valves

- RHR-MO-25B
  - Open to Initiate CSD (and Alt-CSD)
  - Procedure Did Not Identify Number of Contactors
  - Operate Valve Using Hand Wheel
- Remaining CSD Valves Are Not Risk Significant (Redundant Capability)
  - RHR-MO-17, RHR-MO-67, RHR-MO-25A, RR-MO-53A

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Significance Determination Discussion CNS Evaluation

Kent Sutton

Risk Management Supervisor

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Significance Determination Discussion

- Guidance for Evaluation of Fire Risk
- Initial CNS Evaluation
  - CNS PSA-ES083
- New CNS Evaluation Information
  - CNS PSA-ES091
- Summary of New CNS Evaluation Information

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CNS Evaluation
Guidance Available

- NRC PRA Guidance
  - NUREG 2300
  - Regulatory Guide 1.200
- Industry PRA Guidance
  - NUREG 6850
  - ASME PRA Standard
- Existing CNS PRA Model is Appropriate
  - Transition To and Maintaining Cold Shutdown Modeling Not Required
- Focused Fire PRA Developed to Address Issue

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CNS Evaluation
SDP Application

- Use and Application of Significance Determination Process (SDP)
  - Compliance with Appendix R
    - Requires establishing & maintaining CSD
  - SDP
    - Hot Shutdown Is End State
    - Cold Shutdown (Screen Green)
  - IMC 0609, Appendix F
    - IMC 0308, Attachment 3, Appendix F

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Initial CNS Cold Shutdown Evaluation

- Finding Categories That Apply
  - Cold Shutdown
  - Post-Fire Safe Shut Down (SSD)
    - Includes Control Room abandonment
  - Localized Cable or Component Protection
- Initial Qualitative Screening
  - All Categories are Screened
  - **Question 2:** Does the Finding Only Affect the Ability to Reach and Maintain Cold Shutdown Conditions?
- IMC 0308 Provides Basis
  - Internal Event PRA and Fire PRA Use Same End State
  - Other Areas put Emphasis on Hot Shut Down (HSD)
- Therefore CNS Screened All CSD Valves From Further Analysis

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Initial CNS Hot Shutdown Evaluation

- CNS PSA-ES083
  - Based on CNS IPEEE Insights
    - 2006 PRA model results
  - Accrued Risk Due to Potential HPCI Overfill
    - Some detailed circuit analysis used
  - Determined Shutdown Cooling is Not Risk Significant
  - RHR-MO-25B Screened Out for Alternate Shut Down (ASD) Areas
    - No contribution from Control Room abandonment scenarios
- Calculated ΔCDF 4.8E-07, Not Risk Significant

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New CNS Specific Information Evaluation Refinements

- Utilized Industry Fire PRA Methods
  - Industry Developments in Fire Modeling
  - Extent of Fire Damage
  - Control Room Abandonment Frequency
  - Ignition Frequency Calculation
- Incorporated this Information into CNS Significance Evaluation
  - CNS Engineering Study PSA-ES091

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New CNS Specific Information PSA-ES091

- 5.4POST-FIRE, Control Room Response Scenarios
  - Application of NUREG 6850
    - Fire modeling within limitations of NUREG 1824
    - Additional circuit analysis performed
    - New ignition frequencies calculated
  - Evaluated Risk Associated with HPCI Valves
    - HPCI high level trip will be working for 4 of 5 fire scenarios evaluated at Reactor Building 903
    - Confirmed HPCI system can be controlled from Control Room
  - Confirmed Shutdown Cooling is Not Risk Significant

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New CNS Specific Information PSA-ES091

- 5.4FIRE-S/D, Control Room Abandonment Scenarios
  - Application of NUREG 6850
    - Fire modeling within limitations of NUREG 1824
    - New fire severity and probability of non-suppression calculated
    - Evaluation of fire growth and propagation related to source
    - Control Room fire - abandonment due to habitability
    - New ignition frequencies calculated
  - One Valve Affected
    - RHR-MO-25B, cold shutdown function
  - Applied Recovery Actions
    - Includes RHR-MO-25B or HPCI recovery
    - Additional success paths available

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New CNS Specific Information Summary of Results

- 5.4POST-FIRE
  - ΔCDF 7.3E-08
    - HPCI not risk significant when fire modeling used
    - Sensitivity study confirmed SDC not risk significant
- 5.4FIRE-S/D
  - ΔCDF 1.3E-08
    - Limited contribution from Control Room abandonment due to smoke
    - Limited fire scenarios causing Control Room abandonment
- Results Are Appropriate to Address Risk Significance of Identified Procedure Issues

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Operator Response

Rod Penfield
Assistant Operations Manager

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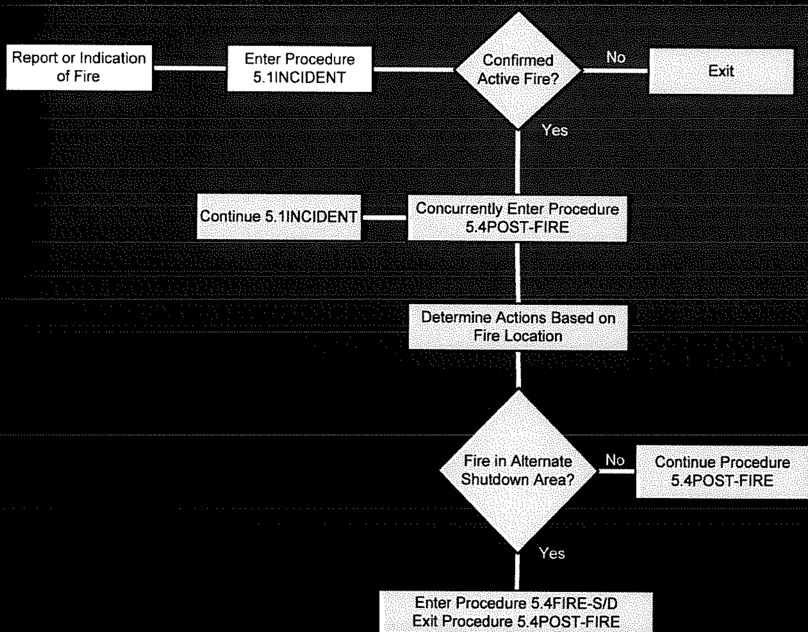

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Procedure Use

- Enter 5.1INCIDENT
  - Dispatch Fire Brigade
  - Enter EAL as Required
- Enter 5.4POST-FIRE
  - Guidance Provided to Address Fire-Related Damage
  - Continue Plant Operations using Normal, Abnormal and EOP Procedures

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Procedure Use (cont.)

- Emergency Procedure 5.4FIRE-S/D
  - Entered Based on Fire Location as Directed by 5.4POST-FIRE
- Control Room Abandonment Based on Either of the Following:
  - Reports of Spurious Operation of Components
  - Control Room Habitability
- If Control Room Abandonment Deemed Necessary, the Following Key Actions take Place:
  - Reactor Scram
  - Close MSIVs
  - ALERT declared

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Procedure Use (cont.)

- Establish and Maintain Hot Shutdown
  - Diesel Generator Supplying Electrical Power
  - Vessel Level and Pressure Maintained with HPCI
    - Interlocks/Automatic features bypassed at ASD panel
  - Suppression Pool Cooling Established

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Procedure Use (cont.)

- Establish and Maintain Cold Shutdown
  - Open 3 Safety Relief Valves
  - Open RHR-MO-25B Locally (at starter)
  - When RPV Pressure Reaches 150 psig, Open RHR-MO-27B
    - Expect RHR Flow and Reactor Level to raise
  - When RPV Level Rises to ~50" or RPV Pressure Lowers to 100 psig, HPCI is Secured
  - Establish Alternate SDC

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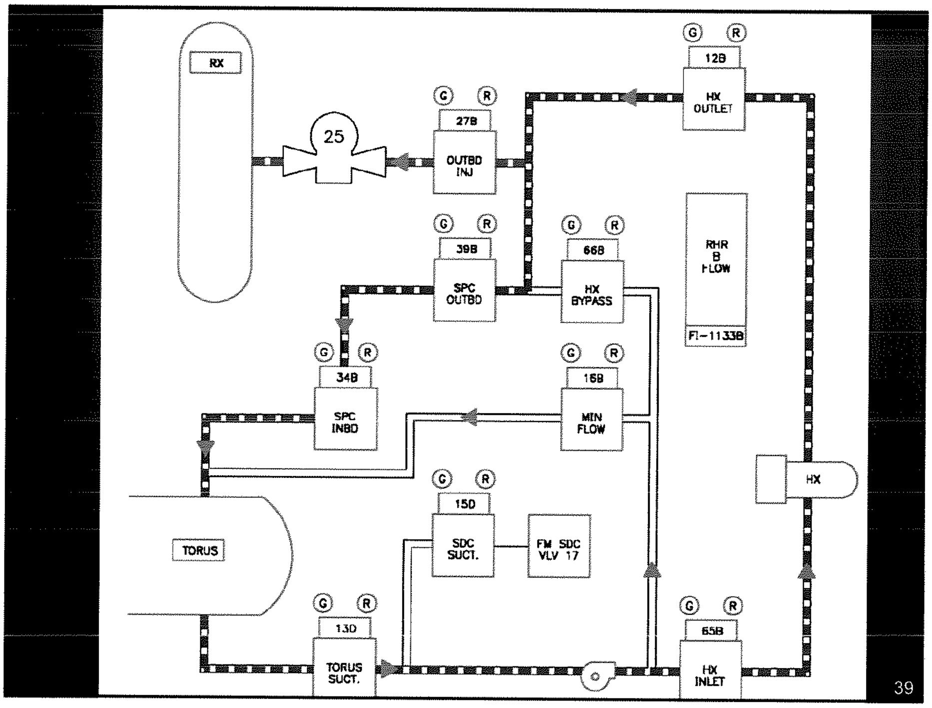

RHR-MO-25B Did Not Open

- ASD Operator Determines that RHR Flow and RPV Level are not Raising (not expected response)
- Observed Symptoms of Problem
  - Operator Determines that RHR is Not Injecting into RPV
  - Operator Observes RHR Suppression Pool Cooling is Operating (as expected)
  - RHR-MO-27B Indicates Full Open
- Evaluate Condition
  - Validate Lineup
  - Check Indications
- Concludes RHR-MO-25B Not Open

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Recovery Actions

- Determine Actions to Establish Flow Through RHR-MO-25B
  - Operator Locally Operates Hand Wheel
  - TSC Sends Personnel to Locally Operate Hand Wheel
- Available Options
  - Re-establish HPCI Operation
    - Guidance for un-isolation and operation of HPCI, from isolated condition, is fully contained within Procedure 5.4FIRE-S/D
  - TSC Develop a Different Injection Source
    - 5.3ALT-STRATEGY (open RHR-MO-25B with hand wheel)

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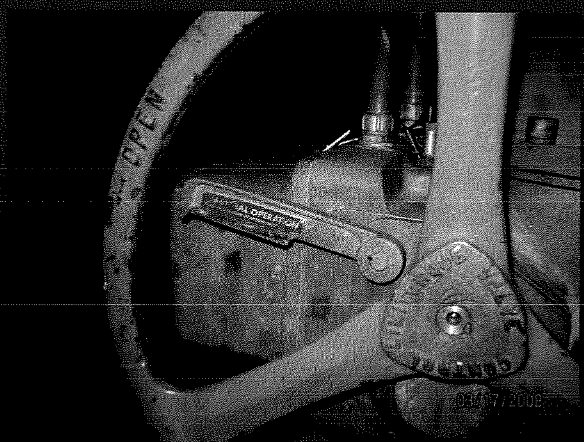
Declutch Lever and Handwheel of RHR-MO-25B



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Significant Determination Discussion Comparison

Kent Sutton

Risk Management Supervisor

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New CNS Specific Information Human Reliability Analysis

- Evaluation of Restoring Reactor Pressure Vessel (RPV) Injection
  - Basis for Recovery Credit
    - Experience and training
    - SRO involved in the diagnosis with necessary indications available at the ASD panel
    - Sufficient time available
- Recovery is Applicable
  - SPAR-H Method (NUREG/CR-6883)
  - Simple and Timely
- Combined Failure to Restore RPV Injection (<4.0E-3)

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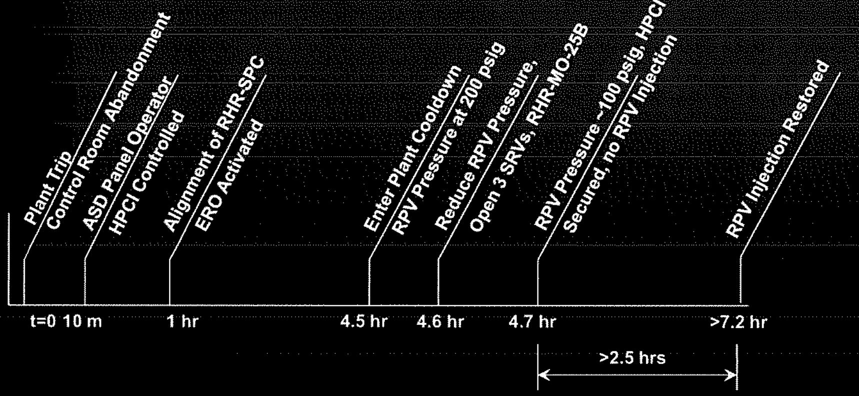
Postulated CR Abandonment Fire

Expected Timeline of Major Events



Limiting time to restore RPV Injection is 2.5 hrs

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Comparison of New CNS Information Evaluation Inputs

Evaluation Input Item		NRC IR 2008-007	CNS PSA-ES091
Fire Ignition Frequency (Excluding Control Room)		IPEEE	NEDC 08-32
Fire Ignition Frequency (Control Room Only)		GENERIC	NEDC 08-32
Control Room Forced Abandonment Frequency		1.42E-05	9.9E-07
Number of Reactor Building 903 MCC Fires Failing HPCI		5	1
Control Room Abandonment Frequency		1.08E-04	1.8E-05
Manual Fire Suppression		2.4E-01	0.8 – 5.E-03
Alternate Shutdown Panel Delta Conditional Core Damage Probability		9.E-01	2.7E-03
Diesel Fire Water Injection Failure Probability (Medium Dependence)		1.0	2.7E-01
HPCI Non-recovery Probability	Operator Actions	1.0	<4.E-03
RHR-25B Non-recovery Probability		1.0	

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NRC SDP Inputs and Assumptions

- NRC IR 2008-007 Uses a Simplified Bounding Approach
  - Limited Use of CNS IPEEE Insights
    - Calculated high Control Room abandonment frequency
    - Inconsistent use of ignition frequencies
  - No Condition Recovery Credit
  - No Plant Specific Fire Modeling
  - Redefines End State for Control Room Abandonment
- Simplified Analysis to Determine Significance Less Than Red

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Key Results of Fire PRA Insights

- CNS Specific Fire Model Results Greatly Reduces Estimated Risk
  - Reduced Fire Ignition Frequencies per NUREG 6850
  - Control Room Abandonment
    - Hot gas layer contribution is minimal
    - No fixed sources in several fire zones
    - Modeling determined low likelihood of forced abandonment
  - Fire Modeling Shows Limited HPCI Impact for Non-Abandonment Scenarios

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Comparison RB 903 SE & SW

- NRC IR 2008-007
  - Mitigated from Control Room
  - ΔCDF = 2.1E-06
- CNS PRA-ES091
  - ΔCDF = 7.3E-08
- Key Factors Affecting Sequences
  - HPCI Isolation Affected by Single 1 Motor Control Center (MCC) Fires in 2A/2C
  - No Impact on HPCI in Zone 2B or Zone 2D
  - Remove all but MCC-Y and ASD Panel Fires in 2A/2C
  - Realistic Fire Growth Outside of Cabinets, 844kW Maximum Size

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Comparison RB 903 NE

- NRC IR 2008-007
  - Control Room Abandonment Frequency, 5.6E-05
  - Base CCDP of 0.1, Case CCDP of 1.0
  - ΔCDF = 5.6E-06
- CNS PRA-ES091
  - Control Room Abandonment Frequency, 5.9E-07
  - ΔCDF = 4.2E-10
- Key Factors Affecting Sequences
  - Smoke and Heat Detectors and Wet Pipe Sprinkler System
  - No Fixed Ignition Source Causes Control Room Abandonment
  - Hot gas layer contribution is minimal
  - Electrical Cabinets are Non-vented and Robustly Secured
  - MCC-K Fire has Limited Impact on Conduits in Zone of Influence

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Comparison
Cable Expansion Room

- NRC IR 2008-007
  - Control Room Abandonment Frequency, 1.7E-06
  - Base CCDP of 0.1, Case CCDP of 1.0
  - ΔCDF = 1.7E-07
- CNS PRA-ES091
  - Control Room Abandonment Frequency, 2.4E-07
  - ΔCDF = 1.7E-10
- Key Factors Affecting Sequences
  - Smoke Detection and Wet Pipe Sprinkler System
  - No Fixed Ignition Sources, and Hot Gas Layer Contribution is Minimal

Comparison
Cable Spreading Room

- NRC IR 2008-007
  - Control Room Abandonment Frequency, 2.5E-06
  - Base CCDP of 0.1, Case CCDP of 1.0
  - ΔCDF = 2.5E-07
- CNS PRA-ES091
  - Control Room Abandonment Frequency, 1.3E-05
  - ΔCDF = 9.3E-09
- Key Factors Affecting Sequences
  - Smoke Detection, Heat Activated Devices and Pre-action Sprinkler
  - Majority of Electrical Panels are Non-vented and Robustly Secured
  - Limited Fixed Sources Contributing to Abandonment Frequency

Comparison
Control Room

- NRC IR 2008-007
  - CR Forced Abandonment Frequency, 1.4E-05
  - Base CCDP of 0.1, Case CCDP of 1.0
  - ΔCDF = 1.4E-06
- CNS PRA-ES091
  - CR Forced Abandonment Frequency, 2.2E-06
  - ΔCDF = 1.6E-09
- Key Factor Affecting Risk
  - Detailed Fire Modeling Shows that the Availability of the Control Room Ventilation System Significantly Reduces Control Room Abandonment Due Habitability Conditions

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Comparison
Auxiliary Relay Room

- NRC IR 2008-007
  - Control Room (CR) Abandonment Frequency, 4.8E-05
  - Base CCDP of 0.1, Case CCDP of 1.0
  - ΔCDF = 4.8E-06
- CNS PRA-ES091
  - Control Room Abandonment Frequency, 2.2E-06
  - ΔCDF = 1.6E-09
- Key Factors Affecting Sequences
  - Smoke Detection System
  - All Cabinets, Except for One, are Non-vented, Robustly Secured
  - Fixed Ignition Source will Not Cause CR Abandonment
  - CR Abandonment Limited to Transient Fires in the Room

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Summary

- NRC Bounding Analysis
  - Fires in Defined Zones Result in Failure of Equipment of Interest
  - Assumes Excessive Conditional Core Damage Probability for CSD Failure
  - CSD Recovery Not Applied
- CNS Focused Fire PRA Evaluation
  - Accounts for Spatial Aspects of Fire Zones
  - Identifies Remaining Mitigation Equipment
  - CSD Does Not Produce a Significant ΔCDF
  - CSD Not Modeled when HSD Shown to be Successful
  - Re-establishment of HSD Not Adversely Affected by CSD Valves
  - Failure to Establish CSD Easily Recoverable
- Differences Are Due to CNS Detailed Analysis

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Closing Remarks

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