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

June 14, 2011

David J. Bannister, Vice President
and Chief Nuclear Officer
Omaha Public Power District
Fort Calhoun Station FC-2-4
P.O. Box 550
Fort Calhoun, NE 68023-0550

SUBJECT: SUMMARY OF MEETING WITH OMAHA PUBLIC POWER DISTRICT REGARDING A PRELIMINARY SUBSTANTIAL FINDING

Dear Mr. Bannister:

This refers to the public regulatory conference conducted in Arlington, Texas on June 2, 2011, between the NRC and your staff. The participants discussed the circumstances associated with a preliminary finding with substantial safety significance regarding the failure of a reactor protection system contactor at the Fort Calhoun Station.

This meeting was classified as a Category 1 public meeting, as communicated in the meeting notice (ADAMS ML111380732). This provided an opportunity for members of the public to discuss regulatory issues with the NRC after the business portion of the meeting, but before the meeting adjourned. No comments were brought forward by the public.

The attendance list for the meeting is enclosed with this summary (Enclosure 1). A copy of the Omaha Public Power District presentation slides is also enclosed (Enclosure 2).

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, and its enclosures, will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records 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,

/RA/

Jeffrey A Clark, Chief
Project Branch E
Division of Reactor Projects

Dockets: 50-285
Licenses: DPR-40

Enclosures:
1. Attendance List
2. OPPD Presentation Slides

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R:\REACTORS\FC MS Reg Conf 6-2-2011 ADAMS ML

ADAMS: □ No ☒ Yes	☒ SUNSI Review Complete	Reviewer Initials: JFM1
	☒ Publicly Available	☒ Non-Sensitive
	□ Non-publicly Available	□ Sensitive
KEYWORD: Fort Calhoun Station Regulatory Conference June 2011		

PE:DRP/PBE	C:DRP/PBE		
JMelfi	JClark		
/RA/	***RVA for JClark***		
06/7/2011	06/14/2011		

OFFICIAL RECORD COPY T=Telephone E=E-mail F=Fax

NRC PUBLIC MEETING ATTENDANCE

LICENSEE/FACILITY	Omaha Public Power District Fort Calhoun Station
DATE/TIME	June 2, 2011; 8:00 a.m.
LOCATION	U.S. Nuclear Regulatory Commission Region IV 612 East Lamar Boulevard, Suite 400 Arlington, Texas 76011-4125

NAME (PLEASE PRINT)	ORGANIZATION
Donna Guinn	OPPD
SUSAN BAUGHN	OPPD
Dave Bannister	OPPD
Jeff Reinhart	OPPD
John Herman	OPPD
Hans Iwand	ESI
Jay Fluehr	Westinghouse
Jerry Shuck	OPPD
RAY KELLAR	RIV NRC
DAVID LOVELESS	RIV
Elmo Collins	RIV
Kriss Kennedy	RIV

NRC PUBLIC MEETING ATTENDANCE

LICENSEE/FACILITY	Omaha Public Power District Fort Calhoun Station
DATE/TIME	June 2, 2011; 8:00 a.m.
LOCATION	U.S. Nuclear Regulatory Commission Region IV 612 East Lamar Boulevard, Suite 400 Arlington, Texas 76011-4125

NAME (PLEASE PRINT)	ORGANIZATION
Jeff Clark	NRC
John Kirkland	NRC
Neil O'Keefe	NRC
Art Howell	NRC
Chris Smith	NRC
SAM GRAVES	NRC
THOMAS R. FARNHOLTZ	NRC
Eduardo R. Uribe	NRC
ZACHARY HOLLCRAFT	NRC
Ramon Azua	NRC

Fort Calhoun Station Operation with a Degraded RPS Contactor

June 2, 2011

Introduction

Jeff Reinhart

Site Vice President

Summary of Differences

- Shorter exposure time (t/2 + repair vs. t + repair)
- Higher Operator Reliability in tripping reactor
- Lower Clutch Power Supply Breaker failure probability
- Common Cause Failure Determination

Agenda

• Risk Assessment	John Herman, DM Engineering
• M Contactor FMEA/CB Breaker Failure Analysis	Hans Iwand, Consulting Engineer
• Finding Significance	Jay Fluehr, Consulting Engineer
• Closing Remarks	Jeff Reinhart, Site VP

Why We Are Safe Today

- 6/16/10 – M2 contactor repaired
- 2/5/11 – M1, M2, M3, M4 contactors replaced
- Operations memorandum on required actions for an inoperable contactor
- Changed the FID level of the M contactors and interposing relays to FID 1 (N1)
- New preventive maintenance procedures
- Briefed FCS leaders on this event to address the behaviors that led to this occurrence

Risk Assessment

John Herman, PE

Division Manager Engineering

Risk Assessment

- Identified the need for additional information
  - Failure modes & effects analysis for FCS contactors
    - Needed for exposure time determination and to evaluate common cause failure criteria
  - Failure analysis data for FCS breakers
    - Needed to determine breaker reliability
  - FCS post accident thermal-hydraulic analysis
    - Needed to determine available time for proper treatment of manual actions

M Contactor FMEA
CB Breaker Failure Analysis

Hans Iwand, PE
Sr. Managing Consultant
ESI-Nebraska

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Failure Analysis

- M contactor multi-disciplined failure analysis
  - Failure analysis on removed RPS M contactors
  - Identify failure modes
    - Electrical and mechanical
  - Evaluate condition of removed M contactors

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Failure Analysis

- Laboratory analysis instructions
  - Identify how the shading coil is retained in the groove.
  - Identify what the material is on the surface, as well as, inside the grooves of the Yoke.
  - Determine the composition of the "dark brown" material found in the bottom of M2 cabinet, as well as, adhered to the M2 components.
  - Inspect the surfaces of the M2 Yoke and Armature to identify the cause of the wear present.
  - Determine the operating temperature of the shading coil.
  - Inspect Contactors M1, M3 and M4 for any signs of wear or degradation.

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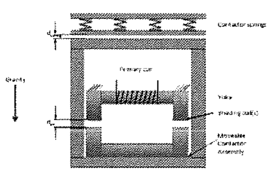
Failure Analysis

Relay Arrangement



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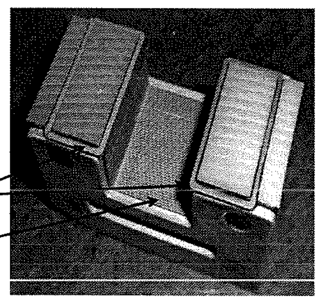
Failure Analysis

Fixed iron core of the contactor with shading coils

Shading coils
•Increase life expectancy and reliability of contactors
•Dampens chattering

Shading coils

Laminated Iron plates



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Failure Analysis

- Failure Modes and Effects Analysis (FMEA)
  - Identify potential failure modes
    - Failure of shading coil
    - Foreign particles
    - Electrical
    - Other
  - Consequence ranking

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Failure Analysis

- Sequence of events
  - Shading coil loose
  - Shading coil drops from groove when contactor cycled
  - Shading coil no longer effective and relay chatters
  - Shading coil spatially constrained between Yoke and Armature
    - Continuously hammered/peened (chatter)
  - Shading coil fractures and fragments escape confines of Yoke and Armature
  - Shading coil fragments prevent contactor from opening

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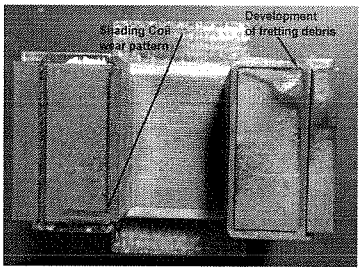
Failure Analysis

Testing

- Manually dislodged shading coil
- Dislodged shading coil will cause contactor to chatter
- Operated 16 hours (chattering)
- Did not cause contactor to become inoperable



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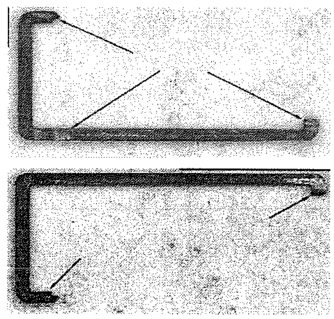
Failure Analysis

Wear condition of failed M2 shading coil

- Red arrows indicate wear conditions and fracture surfaces of both sides of the failed shading coil



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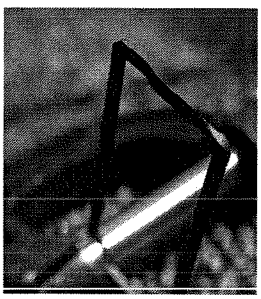
Failure Analysis

Non-fragmented M2 shading coil condition
- Normal shading coil thickness - .0402"
- M2 .0397" - .0065"



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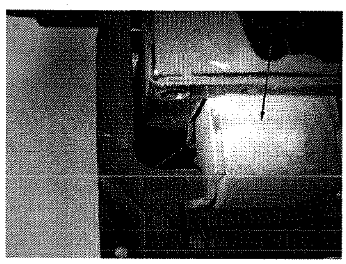
Failure Analysis

M2 Contactor
- Blue arrow – broken shading coil fragments
- Red arrow – wear pattern on Yoke coil



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Failure Analysis

- Conclusion (failure of M2 contactor)
  - Loose shading coils do not cause contactor inoperability
  - Over time chattering contactor has potential to fragment shading coil
  - Whole shading coil will not jam the contactor
  - Failure mode is a heavily worn and fragmented shading coil that jams the contactor

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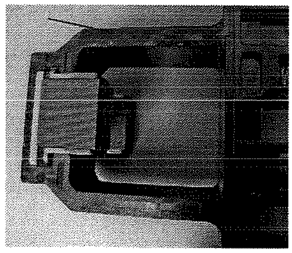
Failure Analysis

M1 contactor wear conditions
- Red arrow points to "indentions" on moveable contact assembly referred to in NRC inspection report



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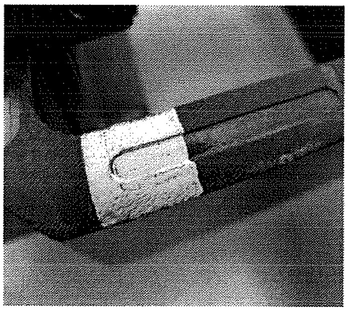
Failure Analysis

M1 contactor wear conditions
- Shows significant "scratching and indentions" referred to in NRC Inspection Report
- Determined to be de-burring operation during manufacturing process
- No effect on operation of the contactor



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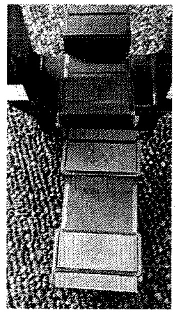
Failure Analysis

M1 Contactor



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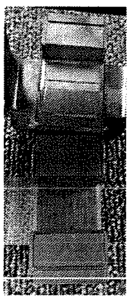
Failure Analysis

M3 Contactor
- Post lab wear testing



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Failure Analysis

M4 Contactor

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Failure Analysis

- Conclusion (condition of other contactors)
  - Inspection of M1, M3 and M4 did not identify any wear or other deleterious conditions similar to M2
  - Failure mode unique to M2 contactor

Failure Analysis

- CB-AB and CB-CD breaker testing
  - Breakers removed from service for testing
  - Breakers tested to failure to determine number of under-voltage trip test cycles
  - CB-AB failed to **reset** after 18,456 cycles
    - Power feed for M1 and M2
  - CB-CD failed to **trip** after 2,187 cycles
    - Power feed for M3 and M4

Finding Significance

Jay Fluehr, PE

Consulting Engineer

Finding Significance

- Shorter exposure time (t/2 + repair vs. t + repair)
- Higher Operator Reliability in tripping reactor
- Lower Clutch Power Supply Breaker failure probability
- Common Cause Failure Determination

Finding Significance

Failed Contactor Exposure Time

Finding Significance

- Assumption 1 of inspection report:
  - Contactor M2 most likely failed on 4/10/10, because vibration during operation insufficient to cause failure
  - Basis for t + repair time
- Based upon ESI report, more appropriate to use t/2 + repair time
- "Risk Assessment of Operational Events Handbook", ("RASP Handbook"), Rev. 1.03, §2.3 provides guidance for exposure time

Finding Significance
RASP Handbook

"For a failure that could have occurred at any time since the component was last functionally operated (e.g., time of actual failure cannot be determined due to the nature of the failure mechanism), the exposure time (T) is equal to one-half of the time period since the last successful functional operation of the component (t/2) plus repair time."

Finding Significance
RASP Handbook

"The t/2 + repair period should be considered for the following cases:
- A thorough root cause assessment by knowledgeable resource experts ruled out failure occurring at the time of the last functional operation, but the inception of the failure after the last operation could not be determined after careful reviews.
- A thorough root cause assessment by knowledgeable resource experts could not rule out the inception of the failure, but a failure mechanism and cause were reasonably known."

Finding Significance

Exposure Time Conclusion
- ESI report demonstrates that RASP Handbook guidance for t/2 + repair is met
- Exposure time (63 days/2 + 1 day to repair) = 32.5 days

Finding Significance

Human Failure Events (HFEs)

Finding Significance

Human Failure Events (HFEs)
- Two operator manual trips in NRC model
  - EOP-00, "Reactor Trip" procedure
  - Differences between OPPD and NRC on available time and dependency
- Additional manual trips in EOP-20, "Functional Recovery", not credited in NRC model
- Determine the time available for success

Finding Significance

Insights from FCS-specific ATWS Analysis

Finding Significance

Historical PRA Treatment of ATWS at FCS

- Detailed RPS models for FCS published in CEN-327-A (1986)
  - Analyses relied on high reliability of RPS design
- Transient analyses based on generic ATWS analyses: conservative but produced acceptable results
- Detailed FCS-specific ATWS analyses not performed at that time because of low risk

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Finding Significance

NUREG-1780 ("Regulatory Effectiveness of the Anticipated Transient Without Scram Rule") ATWS Event Tree

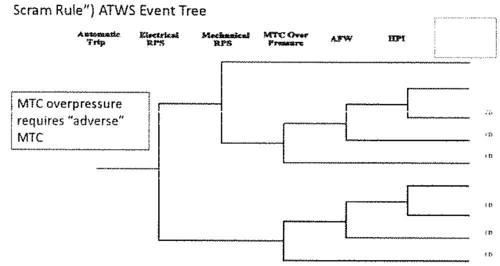

Figure A-4 ATWS Rule Event Tree for Combustion Engineering/ Babcock & Wilcox Reactor Group

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Finding Significance

Period of "Adverse" MTC

- Early RCS pressurization challenges function of ECCS injection valves and other SSCs: irreparable damage
- Unacceptable if pressure exceeds ASME Service Level C, as discussed in NUREG-1780 (3200 psig)
- CE-NPSD-354 (CE DSS Functional Specification) conservatively defines this pressure as 3200 psia

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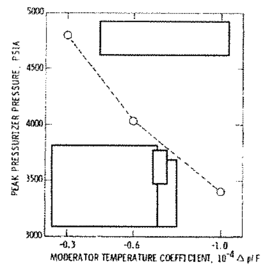
Finding Significance

Peak RCS Pres. v. MTC for Generic 2560 MWT CE PWRs (LOMF Event)



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Finding Significance

ANTICIPATED TRANSIENT		PEAK PRESSURE RESPONSE WITH TYPICAL BOC MTC	
		2560 MWT PWRS	3800 MWT PWRS
1A	CEA Withdrawal from zero power	>4000 PSIA	3800 PSIA
1B	CEA Withdrawal from full power	<2410 PSIA	<2300 PSIA
2	Uncontrolled Boron Dilution	2300 PSIA	3000 PSIA
3	Loss of Coolant Flow	<2320 PSIA	3700 PSIA
4	Idle Loop Startup	2430 PSIA	2500 PSIA
5	Loss of External Load	3430 PSIA	>3800 PSIA
6	Loss of Feedwater	>4000 PSIA	4000 PSIA
7	Loss of Station Power	3000 PSIA	2550 PSIA
8	Excess Load	<2300 PSIA	2500 PSIA
9	RCS Depressurization	<2250 PSIA	<2250 PSIA
10	Primary Sample Line Break	<2250 PSIA	<2250 PSIA

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Finding Significance

FCS-specific ATWS analyses

- M2 contactor issue challenged assumption that RPS reliability would be maintained
- FCS-specific ATWS analyses is needed to accurately determine risk significance
- General model structure consistent with NUREG-1780 event tree

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Finding Significance

FCS-specific ATWS study performed by Westinghouse

- Best estimate CENTS transient analysis code
- Bounding transient events based on CENPD-158-P, "ATWS Analyses: Analysis of Anticipated Transients without Scram for Combustion Engineering NSSS's"

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Finding Significance

- Bounding transient events analyzed
  - Loss of Main Feedwater (LOMF)
    - with and without one closed PORV
    - with and without stuck open PSV
  - Station Blackout (SBO)
  - Loss of Condenser Vacuum (LOCV)
- Small break LOCA added to enhance risk assessment

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Finding Significance

Loss of Main Feedwater (LOMF)

- Most limiting "at power" ATWS event from CENPD-158-P
- Analysis Conservatisms
  - BOC least negative (bounding) cycle 26 MTC
  - No credit for inventory makeup or boration from charging
  - Discharge flow rate for combined PORVs/PSVs

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Finding Significance

Summary of results for LOMF Event with one closed PORV

- Peak RCS Pressure < 3200 psia
- No return to power
- No core uncovery
- No core voiding

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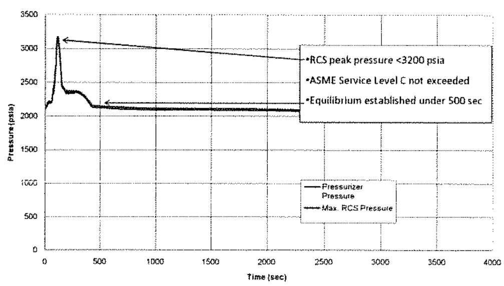
Finding Significance

LOMF with One Closed PORV - RCS Pressure



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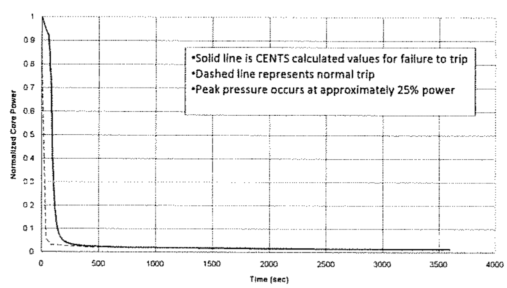
Finding Significance

LOMF with One Closed PORV - Core Power



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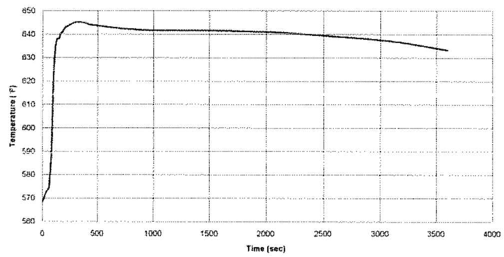
Finding Significance

LOMF with One Closed PORV - Core Average Temperature



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Finding Significance

Summary of Results for Analyzed Transients

- With least negative MTC, RCS pressure < 3200 psia -- margin improves throughout cycle as MTC decreases
- Core remains covered for the one hour duration of T-H runs
- For scenarios with relief valve reseat
  - Inventory loss limited, core remains covered with saturated liquid for a period in excess of one hour
  - Plant is stabilized on pressure plateau below PORV setpoint

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Finding Significance

Analyzed Transients (continued)

- For scenarios with stuck open relief valve
  - Increased inventory loss
  - Depressurization and SI makeup provide:
    - Reactivity control
    - Inventory control
  - Core covered by two-phase mixture

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Finding Significance

Summary of Results for Small Break LOCA

- Peak RCS Pressure < 3200 psia
- No return to power
- Core uncovery initiates after 13 minutes
- AFW and HPSI mitigate event
- Reactivity controlled via core voiding (short term) and HPSI (long term)

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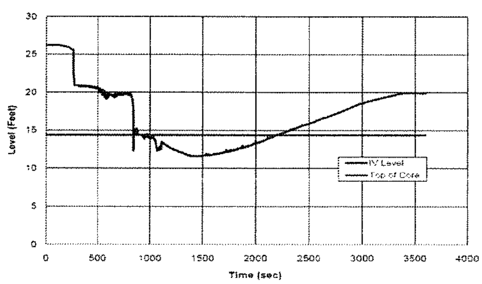
Finding Significance

Small Break LOCA (2" Break) - Inner Vessel Level



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Finding Significance

Application to EOP-00 operator actions

- Implementation before 10 minutes ensures action prior to core uncovery for small LOCAs
- 10 minutes used for PRA-significant transients (e.g., reactor trip, turbine trip, and loss of off-site power) - conservative assumption
- 2 minutes appropriate for other transients (before peak pressure) - conservative assumption and low PRA significance

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Finding Significance

Application to EOP-20 operator actions

- Additional EOP-20 manual trips will be effective prior to core uncovery
- Actions can be implemented for transient events as late as one hour into the event
- Additional EOP-20 actions only applied to PRA significant transients

Finding Significance

End of FCS-specific ATWS Analysis

Finding Significance

1.1 **IF** the reactor did **NOT** trip, **THEN** establish Reactivity Control by performing step a, b, c or d:

a. Manually trip the Reactor (CB-4).

b. Manually trip the Reactor (AI-31).

c. Place the DSS Manual Trip Switches in "TRIP" (AI-66A/B).

d. Manually open the CEDM Clutch Power Supply Breakers (AI-57).

- EOP-00
- Only a and b included in model since c and d also use breakers
- OPD-4-09, "EOP/AOP Users Guidelines": if unable to perform step a, attempt b, etc.

Finding Significance

Video of EOP-00

Operator Actions

Finding Significance

EOP-00 Operator Actions

- First action is RPS-XHE-XM-SCRAM, Operator Fails to Manually Trip the Reactor
- Second action is RPS-XHE-ERROR, Operator Fails to De-energize CEDM Power Supply (Recovery Event)
- As seen in video, steps essentially performed concurrently, by primary and secondary operators respectively

Finding Significance

Operator Fails to Manually Trip the Reactor - RPS-XHE-XM-SCRAM

- Represents failure to de-energize M contactors
- First action by primary operator
- OPPD recommends one change to SPAR-H timing for small LOCAs and risk-significant transients (e.g., reactor trip, turbine trip, and loss of off-site power)
- Human error probability includes available time recommended by SPAR-H

Finding Significance

RPS-XHE-XM-SCRAM

SPAR-H Values for Small LOCAs and Risk-significant Transients		
	Initial NRC Value	Recommended Value
Time until irreversible damage	2 minutes	10 minutes
Human error probability (HEP)	1.5E-03	6.0E-04

Finding Significance

Operator Fails to De-energize CEDM Power Supply (Recovery Event) - RPS-XHE-ERROR

- Represents failure to trip clutch power supply breakers
- Represents four attempts in EOP-00
  - Second action by primary operator
  - First, second, and third actions by secondary operator

Finding Significance

RPS-XHE-ERROR (continued)

- Steps 1.1.b, c, and d in EOP-00
- Per OPD-4-09 "EOP/AOP Users Guidelines," if unable to perform step a, perform step b, etc.
- Operators are trained to continue performing list of steps until success, or until list is exhausted

Finding Significance

RPS-XHE-ERROR (continued)

- Inspection report: HEP = 0.5 because of dependency with RPS-XHE-XM SCRAM
- 0.5 is correct for cutsets with RPS-XHE-XM-SCRAM and RPS-XHE-ERROR
- Otherwise, should be 6.0E-04

Finding Significance

Example of independent RPS-XHE-ERROR

Basic Event	Description	Event Probability
RPS-CPA-CF-CHNLS	CCF of 3 of 4 Analog Core Protection Calc Channels	1.700E-004
RPS-RYT-CF-M12	Common Cause Failure of Contactors M1 and M2	2.400E-006
RPS-XHE-ERROR	Operator Fails to De-Energize CEDM Power Supply (Recovery Event)	~~4.400E-001~~ (should be 6.0E-04)

Finding Significance

Example of dependent RPS-XHE-ERROR

Basic Event	Description	Event Probability
RPS-CBI-CF-ALL	CCF of all Combinations of Bistables	7.700E-007
/RPS-CHN-BP-CHNLA	Channel A in bypass	9.900E-001
/RPS-CHN-TM-CHNLA	Channel A in T&M	9.840E-001
RPS-XHE-XM-SCRAM	Operator Fails to Manually Trip the Reactor	1.000E-002
RPS-XHE-ERROR	Operator Fails to De-Energize CEDM Power Supply (Recovery Event)	4.400E-001

Finding Significance

RPS-XHE-ERROR (continued)

- OPPD recommends one change to SPAR-H timing for small LOCAs and risk-significant transients (e.g., reactor trip, turbine trip, and loss of off-site power)
- OPPD recommends one change to SPAR-H timing for other transients requiring reactor trip
- Human error probabilities include available times recommended by SPAR-H

Finding Significance

RPS-XHE-ERROR

SPAR-H Values for Small LOCAs and Risk-significant Transients		
	Initial NRC Value	Recommended Value
Time until irreversible damage	75 seconds	10 minutes
Dependency	Yes	Only if subsequent to RPS-XHE-XM-SCRAM
Human error probability (HEP)	5.0E-01	6.0E-04

Finding Significance

RPS-XHE-ERROR

SPAR-H Values for Other Transients Requiring Reactor Trip		
	Initial NRC Value	Recommended Value
Time until irreversible damage	75 seconds	2 minutes
Dependency	Yes	Only if subsequent to RPS-XHE-XM-SCRAM
Human error probability (HEP)	5.0E-01	1.5E-03

Finding Significance

*2 Deenergize CEDM(s) by performing step a, b, c, d, e or f:

a. Manually trip the Reactor (CB-4).

b. Manually trip the Reactor (AI-31).

c. Place the DSS Manual Trip Switches in "TRIP" (AI-66A/B).

d. Manually open CEDM Clutch Power Supply Breakers (AI-57).

e. Trip the AI-57 Power Supply Breakers (AI-40A/B/C/D).

f. Open all Individual Rod Drop Test Switches (AI-3).

- EOP-20, RC-1, Step 2
- e and f are additional actions beyond EOP-00

Finding Significance

Video of EOP-20 Operator Actions

Finding Significance

Additional EOP-20 Operator actions

- Independent of M contactors and CB-AB/CB-CD breakers
- Open four 120 VAC breakers from instrument panels, or rod drop test switches
- Actions are on "back panels" in control room
- Time to perform: 15 minutes
- SPAR-H HEP: 1.4E-03

Finding Significance

Breaker Failure Probability

Finding Significance

NRC Generic Breaker Failure Probability

- Used EGG-SSRE-8875, "Generic Component Failure Data Base for Light Water and Liquid Sodium Reactor PRAs"
  - Published in 1990
  - Data source for breakers is Seabrook PRA
- NRC failure probability = holding coil + all other causes = 2.5E-3 + 5.0E-03 = 7.5E-03/demand

Finding Significance

OPPD Generic Breaker Failure Probability

- Used NUREG/CR-6928, "Industry-Average Performance for Components and Initiating Events at U.S. Commercial Nuclear Power Plants"
  - Published in 2007
  - Refers to reactor protection system studies
  - Data source for breakers is NPRDS, 1984-1995
- FCS Failure probability = Mechanical Portion + Under-voltage Device = 1.5E-05 + 4.0E-04 = 4.2E-4/demand

Finding Significance

- Power supply breaker test results
  - Combined 20,643 test cycles
  - CB-AB failed to reset after 18,456 cycles – treated as a failure to trip
  - CB-CD failed to trip after 2,187 cycles
- Methods for estimating plant-specific breaker failure probability
  - Maximum likelihood estimate
  - Bayesian update using informative prior distribution
  - Bayesian update using Jeffreys non-informative prior distribution

Finding Significance

- Bayesian update using Jeffreys non-informative prior distribution was selected
- Appropriate for estimating breaker failure probability using the test results
  - Uncertainty exists in prior evidence (i.e., probability)
  - FCS clutch power supply breakers unique compared with reactor trip breakers at other CE plants
  - Very little prior evidence is conveyed to updated result

Finding Significance

Breaker failure probability using Jeffreys non-informative prior distribution = 1.2E-04/demand

Finding Significance

Summary		
	Initial NRC Value	Recommended Value
M2 Failure Exposure Time	64 days	32.5
RPS-XHE-XM-SCRAM Human Error Probability	1.5E-03	6.0E-04 for small break LOCAs and risk-significant transients
RPS-XHE-ERROR Human Error Probability	0.5	6.0E-04 when independent from RPS-XHE-XM-SCRAM
EOP-20 Actions Human Error Probability	N/A	1.4E-03
Breaker Failure Probability	7.5E-03	1.2E-04
Significance Color	Yellow	Green

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Finding Significance

Common Cause Failure of
M1 Contactor

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Finding Significance

Inspection report, Page A-7,
cites Revision 1.01 of RASP Handbook:

"A component failure should be considered independent (no common cause failure mechanism exists) ONLY when the cause is well understood and there is no likelihood that the same components in other trains or parallel component groups could fail for the same cause. A presumption of zero common cause potential should be a rare occurrence."

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Finding Significance

- These criteria were removed from the current Revision 1.03 of RASP Handbook
- Therefore, FCS consulted criteria in NUREG/CR-5500, *Volume* 10, "Reliability Study: Combustion Engineering Reactor Protection System, 1984-1998," since this document was used to determine CCF value (reference: inspection report, page A-4, item 8)

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Finding Significance

Per NUREG/CR-5500, a CCF event consists of component failures that meet four criteria:

1. two or more individual components fail or are degraded, including failures during demand, in-service testing, or deficiencies that would have resulted in a failure if a demand signal had been received;
2. components fail within a selected period of time, such that success of the probabilistic risk assessment (PRA) mission would be uncertain;
3. component failures result from a single shared cause and coupling mechanism; and
4. component failures are not due to failures of equipment outside the established component boundary.

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Finding Significance

- Guidance for common cause failure (CCF) assessment has changed
- CCF observations in inspection report, p. A-4, assumption 7, may need to be updated based on new information provided in ESI report
- No single clear path for analysis of CCF

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Finding Significance

Sensitivity insights

- Significant impacts on ΔCDP are:
  - Contactor failure exposure time
  - Human error probabilities
  - Breaker failure probabilities
- In aggregate, using the updated information for these areas produces a significance level of GREEN
- Acceptance of the above recommendations would make recalculation of common cause failure unnecessary

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Closing Comments

Jeff Reinhart

Site VP

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Closing Comments

- Shorter exposure time (t/2 + repair vs. t + repair)
- Higher Operator Reliability in tripping reactor
- Lower Clutch Power Supply Breaker failure probability
- Common Cause Failure Determination

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Appendix

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Root Cause Analysis

- Root Cause
  - Key stakeholder (operations, work planning, maintenance and engineering) procedural guidance is insufficient to ensure consistent recognition of nonconforming conditions, resulting in failure to adequately evaluate operability and risk, and influencing restoration decisions based largely on intuitive knowledge and judgment.

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Root Cause Analysis

- Key Contributing Causes
  - Failure to recognize an EQ evaluation was required to reattach the shading coil. (This contributed to subsequent failures.)
  - Fit, form, and function attributes of contactor were not appropriately considered by engineering, maintenance planning, the corrective action program, and the degraded/nonconformance committee.
  - Mindset was that contactor would always perform its intended function to fail open. This desensitized the station to the need for a formal operability determination and risk assessment.
  - Inadequate guidance in the work instructions (EM-RR-RPS-201).
  - Procedures do not require, in all cases, an engineering evaluation be performed for components that are degraded/non-conforming

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Corrective Actions

- Key Corrective Actions to Prevent Recurrence

  Several procedures will be revised to include:
  - Specific instruction to NOT reattach contactor shading coils (EM-RR-RPS-201)
  - Ensure degraded/nonconforming SR components are evaluated for fit, form and function following maintenance activities
  - PRC to verify an engineering evaluation is performed for restoration of degraded/nonconforming conditions
    - Specific Critical characteristics for fit, form, and function
  - Training will be provided to operators, engineers, maintenance craft, planners, and degraded/ nonconforming condition committee members.
  - Perform benchmarking on the tracking and closeout of degraded/nonconforming conditions
  - Perform effectiveness reviews