ATTACHMENT 7
Time Validation – Performance of SBO and Starting of Division 2 Diesel Generator

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Time Validation – Performance of SBO and Starting of Division 2 Diesel Generator

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Time Validation – Performance of SBO and Starting of Division 2 Diesel Generator

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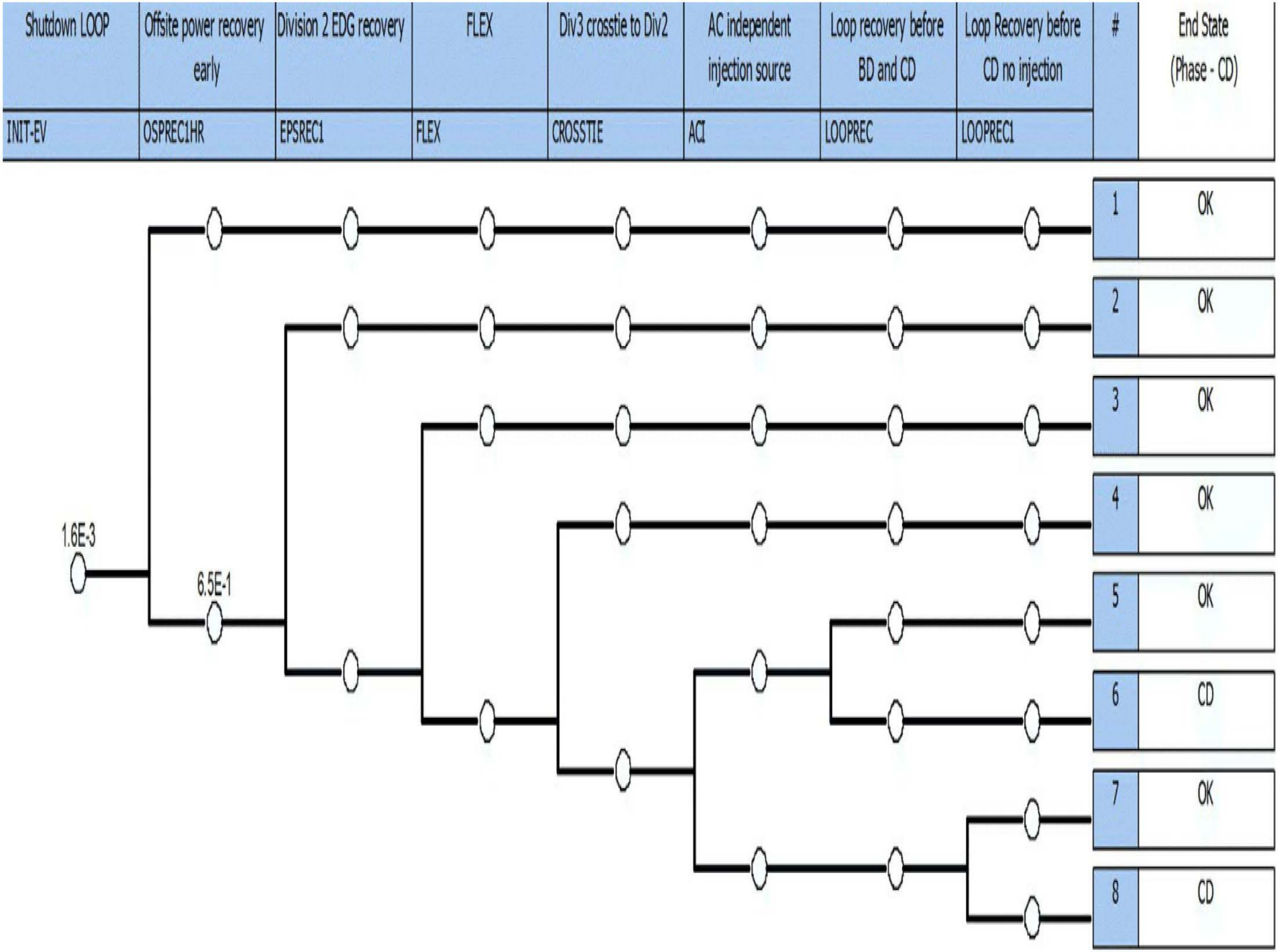
Shutdown LOOP
INIT-EV
Offsite power recovery early
OSPREC1HR
Division 2 EDG recovery
EPSREC1
FLEX
FLEX
Div3 crosstie to Div2
CROSSTIE
AC independent injection source
ACI
Loop recovery before BD and CD
LOOPREC
Loop Recovery before CD no injection
LOOPREC1
#
End State (Phase - CD)
1.6E-3
6.5E-1
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2
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3
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4
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CD
7
OK
8
CD

Cut Set Report - 0-SD-			CLINTON SPAR MODEL		
#	Case	Prob/Freq	Total %	Cut Set	Description

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HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

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HRA Worksheets for LPSD

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Revised SPAR-H Worksheet*

** Revised based on updated guidance provided in INL/EXT-10-18533, Revision 2, "SPAR-H Step-by-Step Guidance."*

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HRA Worksheets for LPSD

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HRA Worksheets for LPSD

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Punch List

Item	Description	Responsible Person	Resolution	Status
1	Change first top event on ET to <DUMMY-FT>	Mitman		Complete
2	There are currently 3 FLEX injection methods. Do I need to add more?	Mitman		
3	Fix SD-CVS ET top event	Mitman		
4	Add manual (is not dependent on electrical power) method to vent containment	Mitman		
5	Should I credit opening primary containment airlock as a method to vent PC?	Mitman		
6	Add method to power SRVs using B5b diesel using CPS 4303.01P004. Modify FT: DEP-SS.	Mitman		
7	Should I credit B5b fire pump as injection method?	Mitman		
8	Should I modify ET to credit low pressure injection without depressurization (prior to boiling)? This is only feasible if procedures direct operator to establish letdown path, which currently we have no evidence of! This would require second set of HEPs with shorter time available.	Mitman		
9	Should offsite power non-recovery probability be based on battery life or 24 hours?	Mitman	(b)(5)	
10	Revisit Div. 2 EDG non-recovery probability	Mitman		
11	Revisit offsite power non-recovery probability	Mitman	[redacted]	
12	Consider solving all ET top event FTs using success criteria	Mitman		Complete for now
13	Find issue with RCIC support system FT	Kozak		
14	Ask Bob Buell to check for model FT renaming errors	Mitman		complete
15	**Sensitivity Cases:**	Mitman		
a	Set HEPs to Exelon values			
b	Decrease HEPs by factor of 0.1			
c	Increased Div. 2 EDG recovery probability			
d	No FLEX credit and non-recovery probabilities based on 24 hours			
e	Case using single dependent HEP for injection methods instead of indep. HEPs			
f				
16	HEP ADS-XHE-XM-MDEPR has a value of 5E-4 from at-power model. Check to see if this is appropriate for SD	Mitman		
17	Compare FLEX DG FS/FR/TM values to Exelon values	Mitman		
18	Compare FLEX diesel driven pump FS/FR/TM values to Exelon values	Mitman		
19	Add HEPs for FLEX disel driven pump transportation and T&M	Mitman		
20	FT: SD-SDC Make sure there is no transfer to Alter SDC which is an artifact of the Grand Gulf model	Mitman		
21	Re-look at HEP times available: My recollection is that TTUC is about 24 hours at low ressure and about 10 hours at high pressure (this time delta makes sense because of the lower heat capacity at ~1000 psig). The implication is that low pressure sequences will have about 24 to core uncovery while high pressure sequences will have half the time.	Mitman		
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Notes

Items	
1	Division 1 electrical system powers outboard containment isolation valves. Div. 2 powers inboard valves.
2	Div. 3 to Div. 2 crosstie: The required lockout resets cannot be performed with AC and DC power (per discussion between SRI and licensee). AC power will be available on Div. 3 if the EDG is running. DC power on Div. 3 should be available. However, DC power will be available on Div. 2 after the Div. 2 battery depletes - this assumes that FLEX electrical has failed.
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ET Top Name	Top Logic	FT* Quantification	Comments
SD-SDC	delete term	1.00	
SD-DEP	system logic	0.31	
SD-LPI	system logic	1.00	
SD-ALT-INJ	delete term	0.39	
SD-HPI	delete term	0.73	
SD-SPC-EXT	delete term	0.64	
SD-ALT-HEAT	delete term	0.01	(b)(5)
SD-CVS	delete term	1.00	
ELEC_XTIE	delete term	0.53	

* all FTs quantified after setting Flag Set = ETF-MF-LOOP

Options on setting the ET Top logic "Process Flag"

Delete Term	
System Logic (I)	
Deveoped Event (W)	(b)(5)

HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

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HRA Worksheets for LPSD

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SPAR HUMAN ERROR WORKSHEET

Laura's HEP

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HRA Worksheets for LPSD

SPAR HUMAN ERROR WORKSHEET

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HRA Worksheets for LPSD

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Laura's HEP

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SPAR HUMAN ERROR WORKSHEET

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HRA Worksheets for LPSD

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Laura's HEP

Note to requester: This document has been withheld in its entirety under FOIA Exemption 5 (deliberative process priviledge).

Human Failure Event (HFE) ID: SD-EPS-XHE-XM-NR01H

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Clinton EDG2 SDP

Risk Summary

11/28/2018
Jeff Mitman

The preliminary best estimate risk evaluation yielded a 3.8E-6 delta core damage frequency which is a white finding. This analysis was calculated using standard and well established NRC probabilistic risk assessment tools and methods and considered the as-built, as-operated plant. We believe this estimate to be realistic. The preliminary risk evaluation considered perspectives provided by Exelon. In addition, sensitivity cases we're performed to evaluate the relationship between key input assumptions and the best estimate risk evaluation. Sensitivity analysis results showed that the finding could range from green to yellow.

The dominant sequence begins with a loss of offsite power followed by: A failure of the Division 2 (b)(5) Emergency Diesel Generator to start [redacted] failure to depressurize the reactor, failure of all high pressure injection systems, and finally, failure to cross tie the functional Division 3 electrical distribution system to the Division 2 system.

(b)(5) [redacted]

We look forward to additional insights from today's discussion. Insights gained will be evaluated to decide whether to perform additional risk analysis. It should be noted that re-evaluation would revisit all assumptions and inputs and that the final risk result may increase or decrease.

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
2443 WARRENVILLE ROAD, SUITE 210
LISLE, ILLINOIS 60532-4352

June 7, 2018

MEMORANDUM TO: Charles Phillips, Project Engineer
Division of Reactor Projects, Branch 1

FROM: Patrick L. Louden, Director ***/RA/***
Division of Reactor Projects

SUBJECT: SPECIAL INSPECTION TEAM CHARTER FOR INOPERABILITY OF THE CLINTON POWER STATION DIVISION 1 AND DIVISION 2 EMERGENCY DIESEL GENERATORS

On Thursday, May 17, 2018, a non-licensed operator discovered the Division 2 emergency diesel generator (EDG) was inoperable due to the air receiver outlet valves being in the closed position. At the time of this discovery, the Division 1 EDG was inoperable and unavailable to complete planned outage maintenance on the Division 1 alternating current (AC) electrical power system. The licensee determined operations personnel failed to open the Division 2 EDG air receiver outlet valves when the EDG was returned to service on May 11, 2018. The licensee subsequently removed the Division 1 EDG from service on May 14, 2018. Once the concurrent inoperability and unavailability of the Division 1 and Division 2 EDGs was identified, operations personnel opened the Division 2 EDG air receiver outlet valves to restore the EDG to an operable status.

The inoperability and unavailability of the Division 1 and Division 2 EDGs resulted in a loss of safety function for the onsite AC electrical power system and placed the Unit in an unplanned shutdown risk red condition for the electrical power key safety function. This condition also caused an unplanned shutdown risk orange condition for the decay heat removal key safety function due to the unavailability of safety-related electrical power to the primary and alternate decay heat removal systems.

Based on the deterministic criteria provided in Management Directive (MD) 8.3, "NRC Incident Investigation Program," the event met MD 8.3 criterion (d), in that there was a loss of safety function for the Division 1 and Division 2 EDGs. The event also met MD 8.3 criterion (h), in that the event raised concerns pertaining to operational performance in the areas of configuration control, risk management and oversight. The risk assessment resulted in an estimated Conditional Core Damage Probability (CCDP) range of E–6 and put the event in the Routine Inspection/Special Inspection overlap region.

In addition to this event, Region III also noted the licensee's recent performance in the areas of configuration control, risk management and oversight. Between May 1 and May 13, 2018, three self-revealing events occurred indicating weaknesses in configuration control, risk management and oversight may be more widespread. The issues included:

CONTACT: Karla Stoedter, DRP
630–829–9731

Inability to Trip a Reactor Recirculation Pump Breaker – During activities to trip the pump breaker, the breaker would not trip. Although the licensee initially believed the breaker's failure to trip was due to circuitry issue, a subsequent review determined licensee personnel were unaware the breaker's control power configuration had changed from energized to de-energized due to planned maintenance on the Division 4 NSPS system.

Unexpected SCRAM Signal due to Maintenance Activities – On May 7, 2018, with one division of the instrument range monitoring system (IRMs) out of service, the licensee performed maintenance and testing on another IRM division which caused a SCRAM signal to be generated. The work instructions in use did not specify an order for disconnecting the test equipment used during this activity. In addition, operations and maintenance personnel did not recognize a SCRAM signal could occur based upon the order the test equipment was removed.

Failure to Verify Valve Position Prior to Operation Results in Equipment Damage – On May 9, 2018, operations personnel directed an equipment operator into the plant to relax the high pressure core spray minimum flow valve off of its seat. Based upon the direction provided, the equipment operator assumed the minimum flow valve was in the closed position and the valve needed to be opened to relax it off its seat. Neither the operations personnel providing the direction nor the equipment operator sent into the plant used configuration control information to validate the minimum flow valve's position. As the operator applied force and attempted to open the valve, the valve was forced into its backseat (due to being in the open position) and over torqued shearing the valve's stem.

Based on the deterministic and risk criteria in MD 8.3, the licensee's recent performance discussed above, and after consultation with NRR, Region III has decided to commence a Special Inspection on June 20, 2018. The Special Inspection will be led by you and will include Robert Murray and Jason Draper. In addition, Laura Kozak, RIII Senior Reactor Analyst, and Jeff Mitman, Senior Reliability and Risk Engineer, will assist the team as needed. The focus of the inspection is to gather information to determine the cause of the EDG event, understand the increased plant shutdown risk condition, and evaluate the licensee's immediate and planned corrective actions for the personnel and process weaknesses that led to the event. On a daily basis, the team should evaluate the need for increasing the scope of the inspection if conditions warrant.

The Team's charter is enclosed.

Docket No. 50–461
License No. NPF–62

Enclosure: Clinton Special Inspection Team Charter

CONTACT: Karla Stoedter, DRP
630–829–9731

DRAFT CLINTON SPECIAL INSPECTION TEAM CHARTER

This special inspection team is chartered to assess the circumstances surrounding the concurrent inoperability and unavailability of the Division 1 and Division 2 emergency diesel generators (EDGs) during the 2018 Refueling Outage. The Special Inspection will be conducted in accordance with Inspection Procedure 93812, “Special Inspection.” The special inspection will include, but is not limited to, the items listed below. This charter may be revised based on the results and findings of the inspection. The inspection results will be documented in NRC Inspection Report 2018050.

1. Develop a complete sequence of events related to the inoperability and unavailability of the Division 1 and Division 2 AC power systems from May 9 through May 17, 2018. The chronology should include plant mode changes, changes in the electrical power, decay heat removal and inventory control shutdown safety/risk areas. Draper, I’d like to include the thought processes of those involved in some of the decision making.

2. Understand the increased shutdown risk condition which existed when no emergency AC power sources were available for a period of approximately 3.5 days. Review the planned shutdown safety configuration compared to the actual configuration that existed. Understand the licensee’s ability to respond to and mitigate a loss of offsite power event given the unavailability of both onsite emergency AC power sources. Murray

3. Review the licensee’s cause analysis efforts and determine if the evaluation’s level of detail is commensurate with the significance of the problem. Phillips

4. Determine the probable cause(s) for the unavailability of the Division 1 and Division 2 EDGs during the 2018 refueling outage. All

5. Understand whether there were any deficiencies in operator training (both licensed and non-licensed operators) which contributed to the EDG unavailability and the failure to identify the condition across multiple operating shifts. Murray

6. Evaluate the licensee’s compliance with, and adequacy of, procedural guidance for performing system alignments, controlling equipment configuration, performing equipment tag-outs and control room log keeping as it pertains to the cause(s) of the event. Draper

7. Evaluate licensee planned and completed corrective actions following the EDG event to the extent possible and assess if prior opportunities (e.g., surveillances, maintenance, and self or nuclear oversight assessments) existed to have identified the problem at an earlier point in time. Murray

8. Determine whether recent internal and external operating experience involving configuration control, risk management and oversight of activities were appropriately evaluated and determine the adequacy of any corrective actions planned or completed. Phillips

9. Continually evaluate the complexity and significance of the event to determine if the circumstances warrant escalation of the inspection to an augmented inspection team. Phillips

10. Identify any lessons learned from the Special Inspection, and prepare a feedback form on recommendations for improving reactor oversight process (ROP) baseline inspection procedures. All

Special Inspection Team

Charles Phillips, Project Engineer, DRP, Special Inspection Team Leader
Robert Murray, Senior Resident Inspector, Quad Cities
Jason Draper, Health Physicist, DNMS

Charter Approval

/RA/	***6/5/18***	K. Stoedter, Chief, Branch 1, Division of Reactor Projects
/RA/	***6/6/18***	K. O'Brien, Director, Division of Reactor Safety
/RA/	***6/7/18***	P. Louden, Director, Division of Reactor Projects

ADAMS Accession Number: ML18158A170

U.S.NRC
UNITED STATES NUCLEAR REGULATORY COMMISSION
Protecting People and the Environment

Phase 3 Risk Assessment
Clinton Emergency Diesel Generator Division 2 Issue
Clinton Units 1

Revision 0.0

DRAFT

Probabilistic Risk Assessment (PRA) Analyst:	Jeff Mitman, Senior Reliability and Risk Analyst, NRR/DRA/APOB
Region III Peer Reviewer	Laura Kozac, Senior Reactor Analyst

File Name: 8e1003a1-9069-4eb8-be32-58ccc50d01b5.pdf
3/5/2020

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MD 8.3 Evaluation
Decision Documentation for Reactive Inspection
(Deterministic and Risk Criteria Analyzed)

PLANT: Clinton	EVENT DATE: 05/11/2018	DETERMINISTIC CRITERIA EVALUATION DATE: 5/18/2018

Brief Description of the Significant Operational Event or Degraded Condition:

On May 9 at 1725, a clearance order for the Division 2 emergency diesel generator (EDG) was removed following a Division 2 bus outage. This clearance order directed the Division 2 EDG air reservoir outlet valves remain closed to prevent the Division 2 EDG from starting since the safety-related service water to the EDG remained out of service. Restoration of the Division 2 EDG and the reservoir isolation valves was tracked via a control room log entry. On May 11, the service water system was restored and the Division 2 EDG was declared available with operability occurring on May 12. Two days later, the Division 1 EDG was declared inoperable for planned maintenance. On May 17, an equipment operator discovered the Division 2 EDG had not been appropriately returned to an available and operable status because the air reservoir outlet valves remained in the closed position. This resulted in the licensee being in Mode 5 and Mode 4 without an operable EDG and a licensee unplanned red shutdown safety condition.

Y/N	DETERMINISTIC CRITERIA
N	a. Involved operations that exceeded, or were not included in the design bases of the facility
	Remarks:
N	b. Involved a major deficiency in design, construction, or operation having potential generic safety implications
	Remarks:
N	c. Led to a significant loss of integrity of the fuel, primary coolant pressure boundary, or primary containment boundary of a nuclear reactor
	Remarks:
Y	d. Led to the loss of a safety function or multiple failures in systems used to mitigate an actual event
(b)(5)	Remarks: [redacted]
(b)(5)	[redacted]
N	e. Involved possible adverse generic implications
	Remarks:

N	f. Involved significant unexpected system interactions
	Remarks:
N	g. Involved repetitive failures or events involving safety-related equipment or deficiencies in operations
	Remarks:
Y	h. Involved questions or concerns pertaining to licensee operational performance
	(b)(5) Remarks: [redacted] (b)(5)

CONDITIONAL RISK ASSESSMENT	
RISK ANALYSIS BY: L. Kozak	RISK ANALYSIS DATE: May 18, 2018

Brief Description of the Basis for the Assessment (may include assumptions, calculations, references, peer review, or comparison with licensee's results):

(b)(5)

The following assumptions were made:

(b)(5)

(b)(5)

The estimated conditional core damage probability (CCDP) is _E-6___ and places the risk in the range of a special inspection and no additional inspection.

RESPONSE DECISION

USING THE ABOVE INFORMATION AND OTHER KEY ELEMENTS OF CONSIDERATION AS APPROPRIATE, DOCUMENT THE RESPONSE DECISION TO THE EVENT OR CONDITION, AND THE BASIS FOR THAT DECISION

DECISION AND DETAILS OF THE BASIS FOR THE DECISION:

BRANCH CHIEF: Karla Stoedter	DATE:
SRA: Laura Kozak	DATE:
DIVISION DIRECTOR: Patrick Louden	DATE:

DIVISION DIRECTOR: Kenneth O'Brien	DATE:
ADAMS ACCESSION NUMBER: EVENT NOTIFICATION REPORT NUMBER (as applicable): EN 53409	

Decision Documentation for Reactive Inspection
(Deterministic-only Criteria Analyzed)

PLANT: Clinton	EVENT DATE: 5/11/2018	EVALUATION DATE: 5/18/2018

Brief Description of the Significant Operational Event or Degraded Condition: On May 9 at 1725, a clearance order for the Division 2 emergency diesel generator (EDG) was removed following a Division 2 bus outage. This clearance order directed the Division 2 EDG air reservoir outlet valves remain closed to prevent the Division 2 EDG from starting since the safety-related service water to the EDG remained out of service. Restoration of the Division 2 EDG and the reservoir isolation valves was tracked via a control room log entry. On May 11, the service water system was restored and the Division 2 EDG was declared available with operability occurring on May 12. Two days later, the Division 1 EDG was declared inoperable for planned maintenance. On May 17, an equipment operator discovered the Division 2 EDG had not been appropriately returned to an available and operable status because the air reservoir outlet valves remained in the closed position. This resulted in the licensee being in Mode 5 and Mode 4 without an operable EDG and a licensee unplanned red shutdown safety condition.

REACTOR SAFETY

Y/N	IIT Deterministic Criteria
N	Led to a Site Area Emergency
	Remarks:
N	Exceeded a safety limit of the licensee's technical specifications
	Remarks:

N	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks:
Y/N	**SI Deterministic Criteria**
N	Significant failure to implement the emergency preparedness program during an actual event, including the failure to classify, notify, or augment onsite personnel
	Remarks:
Y	Involved significant deficiencies in operational performance which resulted in degrading, challenging, or disabling a safety system function or resulted in placing the plant in an unanalyzed condition for which available risk assessment methods do not provide an adequate or reasonable estimate of risk.
(b)(5)	Remarks: [redacted]
(b)(5)	[redacted]

	RADIATION SAFETY
Y/N	**IIT Deterministic Criteria**
N	Led to a significant radiological release (levels of radiation or concentrations of radioactive material in excess of 10 times any applicable limit in the license or 10 times the concentrations specified in 10 CFR Part 20, Appendix B, Table 2, when averaged over a year) of byproduct, source, or special nuclear material to unrestricted areas
	Remarks:
N	Led to a significant occupational exposure or significant exposure to a member of the public. In both cases, "significant" is defined as five times the applicable regulatory limit (except for shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks:

N	Involved the deliberate misuse of byproduct, source, or special nuclear material from its intended or authorized use, which resulted in the exposure of a significant number of individuals
	Remarks:
N	Involved byproduct, source, or special nuclear material, which may have resulted in a fatality
	Remarks:
N	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks:
Y/N	**AIT Deterministic Criteria**
N	Led to a radiological release of byproduct, source, or special nuclear material to unrestricted areas that resulted in occupational exposure or exposure to a member of the public in excess of the applicable regulatory limit (except for shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks:
N	Involved the deliberate misuse of byproduct, source, or special nuclear material from its intended or authorized use and had the potential to cause an exposure of greater than 5 rem to an individual or 500 mrem to an embryo or fetus
	Remarks:
N	Involved the failure of radioactive material packaging that resulted in external radiation levels exceeding 10 rads/hr or contamination of the packaging exceeding 1000 times the applicable limits specified in 10 CFR 71.87
	Remarks:
N	Involved the failure of the dam for mill tailings with substantial release of tailings material and solution off site
	Remarks:

Y/N	SI Deterministic Criteria
N	May have led to an exposure in excess of the applicable regulatory limits, other than via the radiological release of byproduct, source, or special nuclear material to the unrestricted area; specifically • occupational exposure in excess of the regulatory limits in 10 CFR 20.1201 • exposure to an embryo/fetus in excess of the regulatory limits in 10 CFR 20.1208 • exposure to a member of the public in excess of the regulatory limits in 10 CFR 20.1301
	Remarks:
N	May have led to an unplanned occupational exposure in excess of 40 percent of the applicable regulatory limit (excluding shallow-dose equivalent to the skin or extremities from discrete radioactive particles)
	Remarks:
N	Led to unplanned changes in restricted area dose rates in excess of 20 rem per hour in an area where personnel were present or which is accessible to personnel
	Remarks:
N	Led to unplanned changes in restricted area airborne radioactivity levels in excess of 500 DAC in an area where personnel were present or which is accessible to personnel and where the airborne radioactivity level was not promptly recognized and/or appropriate actions were not taken in a timely manner
	Remarks:
N	Led to an uncontrolled, unplanned, or abnormal release of radioactive material to the unrestricted area • for which the extent of the offsite contamination is unknown; or, • that may have resulted in a dose to a member of the public from loss of radioactive material control in excess of 25 mrem (10 CFR 20.1301(e)); or, • that may have resulted in an exposure to a member of the public from effluents in excess of the ALARA guidelines contained in Appendix I to 10 CFR Part 50
	Remarks:
N	Led to a large (typically greater than 100,000 gallons), unplanned release of radioactive liquid inside the restricted area that has the potential for ground-water, or offsite, contamination
	Remarks:

N	Involved the failure of radioactive material packaging that resulted in external radiation levels exceeding 5 times the accessible area dose rate limits specified in 10 CFR Part 71, or 50 times the contamination limits specified in 49 CFR Part 173
	Remarks:
N	Involved an emergency or non-emergency event or situation, related to the health and safety of the public or on-site personnel or protection of the environment, for which a 10 CFR 50.72 report has been submitted that is expected to cause significant, heightened public or government concern
	Remarks:

SAFEGUARDS/SECURITY	
Y/N	**IIT Deterministic Criteria**
N	Involved circumstances sufficiently complex, unique, or not well enough understood, or involved safeguards concerns, or involved characteristics the investigation of which would best serve the needs and interests of the Commission
	Remarks:
N	Failure of licensee significant safety equipment or adverse impact on licensee operations as a result of a safeguards initiated event (e.g., tampering).
	Remarks:
N	Actual intrusion into the protected area.
	Remarks:
Y/N	**AIT Deterministic Criteria**
N	Involved a significant infraction or repeated instances of safeguards infractions that demonstrate the ineffectiveness of facility security provisions
	Remarks:
N	Involved repeated instances of inadequate nuclear material control and accounting provisions to protect against theft or diversions of nuclear material
	Remarks:
N	Confirmed tampering event involving significant safety or security equipment
	Remarks:
N	Substantial failure in the licensee's intrusion detection or package/personnel search procedures which results in a significant vulnerability or compromise of plant safety or security
	Remarks:

Y/N	SI Deterministic Criteria
N	Involved inadequate nuclear material control and accounting provisions to protect against theft or diversion, as evidenced by inability to locate an item containing special nuclear material (such as an irradiated rod, rod piece, pellet, or instrument)
	Remarks:
N	Involved a significant safeguards infraction that demonstrates the ineffectiveness of facility security provisions
	Remarks:
N	Confirmation of lost or stolen weapon
	Remarks:
N	Unauthorized, actual non-accidental discharge of a weapon within the protected area
	Remarks:
N	Substantial failure of the intrusion detection system (not weather related)
	Remarks:
N	Failure to the licensee's package/personnel search procedures which results in contraband or an unauthorized individual being introduced into the protected area
	Remarks:
N	Potential tampering of vandalism event involving significant safety or security equipment where questions remain regarding licensee performance/response or a need exists to independently assess the licensee's conclusion that tampering or vandalism was not a factor in the condition(s) identified
	Remarks:

RESPONSE DECISION
USING THE ABOVE INFORMATION AND OTHER KEY ELEMENTS OF CONSIDERATION AS APPROPRIATE, DOCUMENT THE RESPONSE DECISION TO THE EVENT OR CONDITION, AND THE BASIS FOR THAT DECISION
DECISION AND DETAILS OF THE BASIS FOR THE DECISION:

BRANCH CHIEF: Karla Stoedter	DATE:
SRA: Laura Kozak	DATE:
DIVISION DIRECTOR: Patrick Louden	DATE:
DIVISION DIRECTOR: Kenneth O'Brien	DATE:
ADAMS ACCESSION NUMBER: EVENT NOTIFICATION REPORT NUMBER (as applicable):	

Distribution: (to be inserted by division/branch secretaries)

Clinton Shutdown Model Assumptions

- Time to boil (TTB) = 4.1 hours, based on Exelon document CL-SDP-010 Rev. 1
- Time to core uncover (TTCU) = 24 hours if reactor is maintained atmospheric 10 hours if pressure rises to safety relief valve lift setpoints, based on Exelon document CL-SDP-010 Rev. 1
- Core uncovery is the normal at-power surrogate for core damage. During shutdown, core damage is expected between 1/3 and 2/3 core height which is somewhat after core uncovery, therefore, using core uncovery as a surrogate for core damage is conservative.
- <u>Unavailable and non-recoverable equipment</u> due to test and maintenance (T&M):
  - EDG 1A (note 4160v AC bus 1A1 is energized and available as long as offsite power is available – because the associated EDG is unavailable, this bus will de-energize on loss of offsite power)
  - 480v AC bus 1A
  - 480v AC bus A
  - NSPS 120v Power distribution panel bus A
  - 125v DC battery charger 1A (which is feed from aux. building MCC 1A1)
- Assumed <u>available</u> equipment
  - 480v AC aux. building bus 1L
  - 480v AC aux. building bus 1M
  - 480v AC aux. building bus 1D
  - 480v AC aux. building bus 1E (feed to 125v DC battery charger 1F)
  - 125v DC battery charger 1F (feed from 480v AC aux. building bus 1E)
- According to drawing E02-1DC06 (125v DC & uninterruptible power supply systems) the normal feed to 125v DC bus 1A is via battery charger 1A. Per the licensee, this battery charger was not available due to T&M. Per the same drawing, the backup supply to 125v DC bus 1A is from swing battery charger 1DC11E from aux. building 480v MCVC 1E (1AP28E). Per drawing E02-1AP03 (electrical load diagram), the 1DC25E battery charger is on 480v AC aux. building bus 1D (1AP14E), [redacted] (b)(5) (b)(5)
- The at-power Clinton SPAR model has basic event (BE) failure probabilities for many of the valves that need to be manipulated by the FLEX procedures. These BE failure probabilities are based on data which include failure to open or close based on AC or DC power being available to operate the valve. During the ELAP condition, electrical power may or may not be available to operate the valve. [redacted] (b)(5) (b)(5) (b)(5) Without electrical power these contributions to the failure probability are not possible. This should lower the valve failure probabilities in the model. However, the valves can still be opened manually by an equipment operator (EO) at the valve operator without electrical power. [redacted] (b)(5) (b)(5)

Model Changes

Event Name	Change	Basis for Change	
ET: SD-M4L-LOOP	Recovery times	Recovery of offsite power changed from 18 hours to 24 hours based on Exelon calc.	
FT: SD-AC-REC-24H	Deleted offsite recovery BE for 18 hours and added BE for 24 hours	Recovery of offsite power changed from 18 hours to 24 hours based on Exelon calc.	
FT: SD-AC-REC-24H	Deleted BE EDG recovery in 18 hours and added BE EDG recovery in 1 hour	Div. 2 EDG can only be recovered in 1 hour. After 1 hour, the ELAP procedure removes DC control power from the EDG after which the EDG is not recoverable.	
FT: FLEX-ELEC	Added FLEX Electrical System	Includes HEP, FLEX DG and FLEX bus failures	
FT: DCP-125V-1A-LT	Modified CLINTON DIVISION I 125 VDC POWER IS UNAVAILABLE	Modified to add FLEX Electrical as a means of powering DC	
FT: DCP-125V-1B-LT	CLINTON DIVISION II 125 VDC POWER IS UNAVAILABLE	Modified to add FLEX Electrical as a means of powering DC	
BE: DCP-BCH-TM-1A	Added: DIVISION I 125VDC BATTERY CHARGER in Test and Maintenance	Initially set to default template ZT-BCH-TM with value of 2E-3, but set to True in change set because it was out for maintenance during window of interest	
FT: SD-SPC-EXT	Modified: Added capability to perform SPC using FLEX	Procedure CPS 4306.01P003	
FT: FSF	Added: FLEX Suppression Pool Cleanup and Transfer	Procedure CPS 4306.01P003	
BE: HCS-XHE-XR-MDP	Added: Operator Fails to Recover HPCS Pump after Maintenance	Assigned failure probability of 90%	

High level guidance provided by 4306.01P017

Strategy	Support	Core Cooling	Containment	Spent Fuel Pool
1	• Lineup FLEX generator to Division 1 or 2 480 VAC per 4306.01P001 FLEX Electrical Connections • Lineup FLEX pump to restore Div 1 or 2 SX per 4306.01P002 FLEX UHS Water Supply	• Run available ECCS waterleg pumps • Pressurize and run RCIC • Makeup to RPV per 4306.01P004 FLEX Low Pressure RPV Makeup • Open an SRV	• Set Primary Containment • Establish suppression pool cooling per 4306.01P003 FLEX Suppression Pool Cooling.	Same as Mode 1,2,3 (makeup, ventilate)

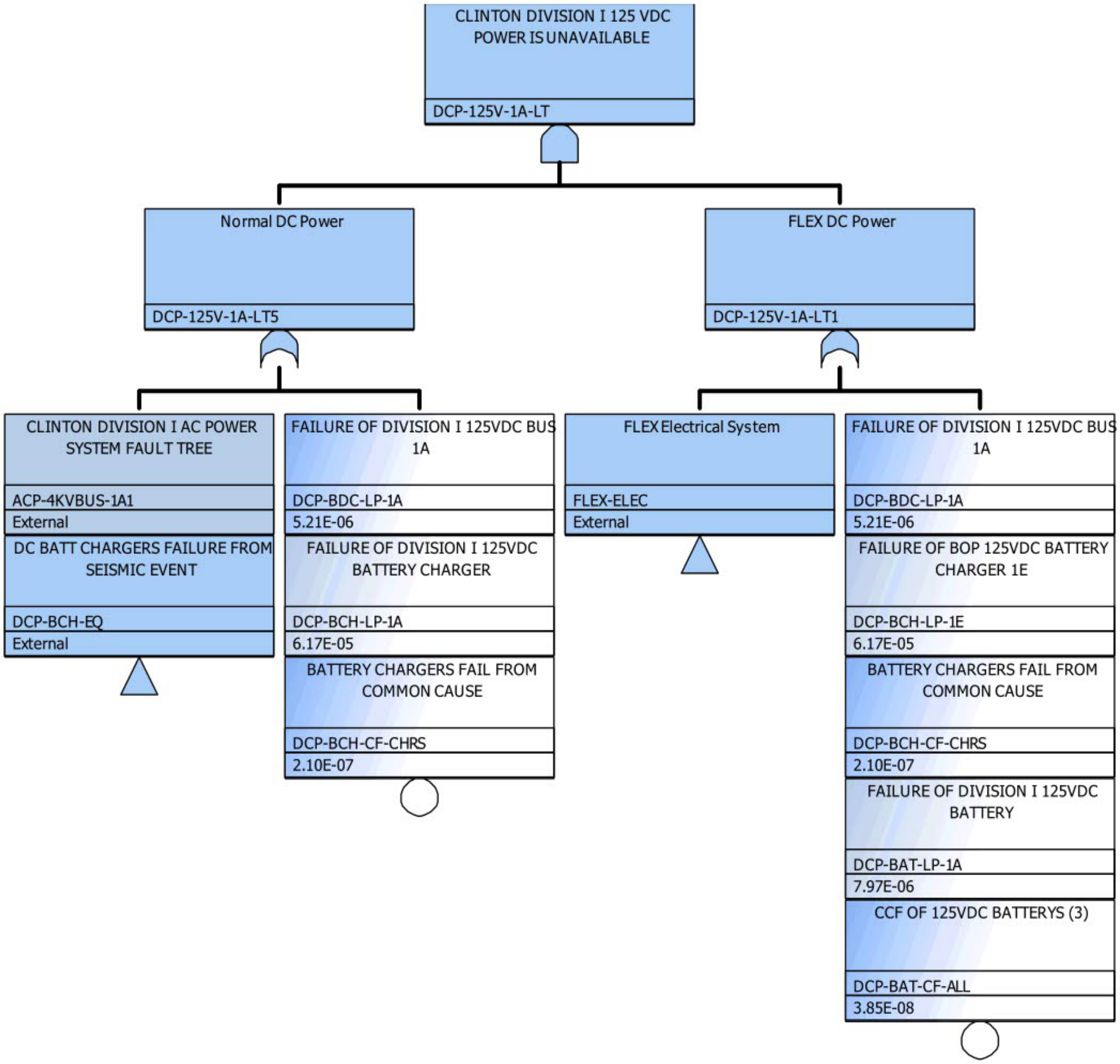
Procedure List

Procedure Number	Title	Current Revision	Revision Controlling during PD	Comments
4006.01	Loss of SDC	5c	5c	
4200.01	Loss of AC Power	25a	25a	
4200.01C002	DC Load Shed during a SBO	5a	5a	
4303.01P023	Cross-Connecting Div. 3 DG to Div1(2)ECCS Electrical Busses	2b	2b	
4306.01P001	FLEX Electrical Connections	0d	0d	Directors operator to DC load shed per 4200.01C002
4306.01P002	FLEX UHS Water Supply	0e	0e	Takes about 6 hours to perform per 4306.01P017
4306.01P003	FLEX SPC			
4306.01P004	Makeup to RCS			Strategy 1 (to be used in CSD) below steps are stipulated in 4306.01P017: • Run ECCS waterleg pumps • Pressurize RCS and run RCIC • Makeup to using this procedure • Open SRV
4306.01P017	ELAP During Modes 4 and 5	0	0	Supplies high level guidance. Including: • "Take action to establish primary containment integrity." • Consider using ECCS waterleg pumps for RPV injection using power from FLEX DG Supplies guidance on using 3 strategies depending on POS. For CSD, directs operator to Strategy 1 (see table above)
4411.06	Emergency Containment Venting, Purging, and Vacuum Relief	6b 8/5/18	??	

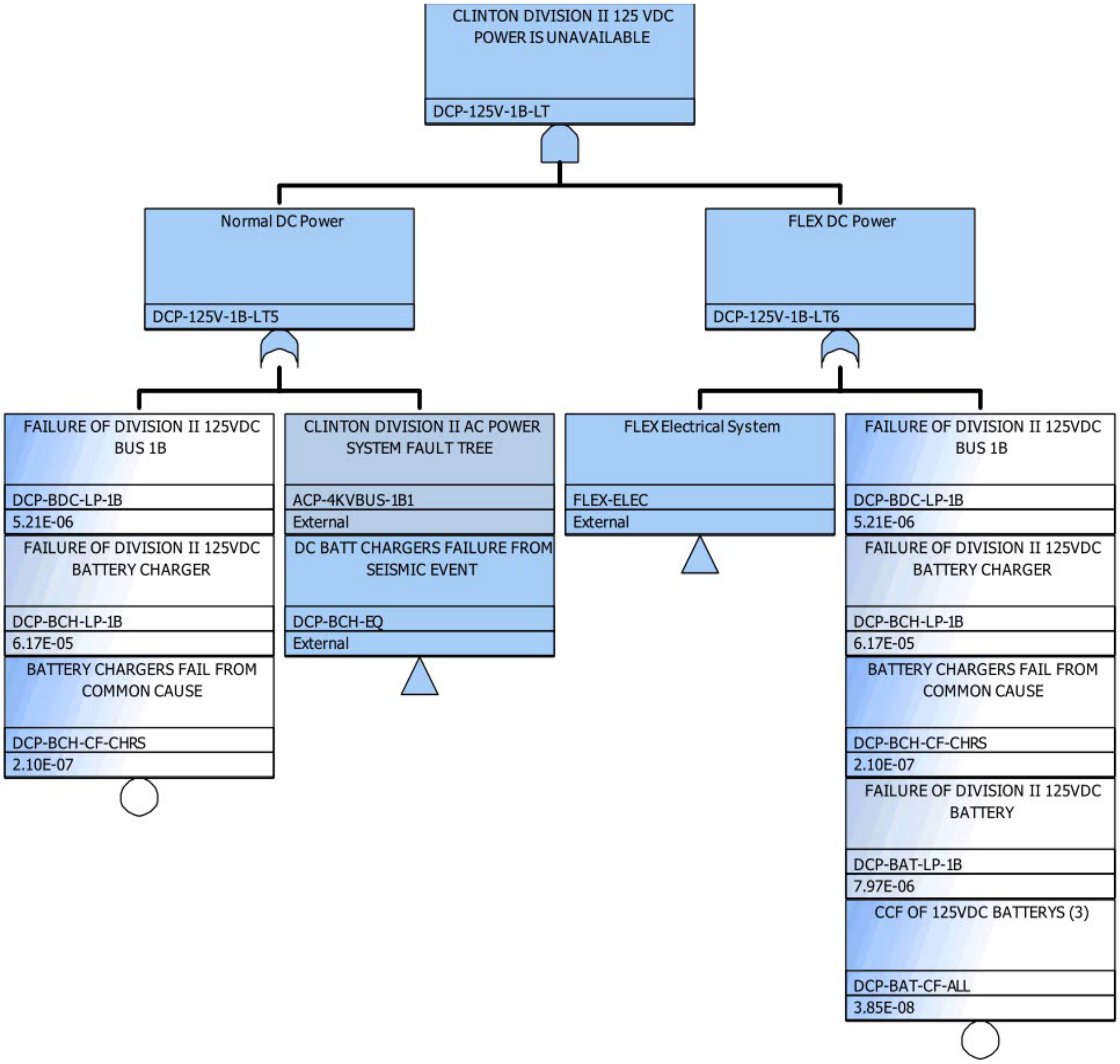
Table x
Summary of Dominant HRA Results

Human Error Event	Description	Procedure	Time Needed	Time Available	Mean Diagnosis HEP	Mean Action HEP	Total Mean HEP
SD-XHE-XM-XTIE	Cross Tie Div. 3 and Div. 2 Electrical	4303.01P023			4.0E-2	6.0E-1	6.4E-1
SD-XHE-XM-FELEC	Operator Fails to Setup and Run FLEX DG and Electrical Distribution	4306.01P001			2.0E-2	2.3E-1	2.5E-1
SD-XHE-XM-FUHS	Operator Fails UHS Water Supply using FLEX	4306.01P002			2.0E-3	1.1E-1	1.1E-1
SD-XHE-XM-FSPC	Operator Fails Suppression Pool Cooling using FLEX	4306.01P003			1.0E-3	2.3E-1	2.3E-1
SD-XHE-XM-FRCS	Injection into RCS using FLEX Diesel Driven Pumps (4306.01P002 Sections 4.3 and 4.4)	4306.01P004	6 hours		2.0E-3	1.1E-1	1.1E-1
SD-XHE-XM-DCLS	Operator performs DC Load Shed	4200.01C002	1hour		4.0E-2	2.0E-2	6.0E-2
SD-XHE-XM-FWS	Operator Fails to Perform Firewater Injection into RCS	?	4 hours	10 hours			1.2E-1
SD-XHE-XM-FRCIC	Operator Fails to Operate RCIC during ELAP from Shutdown	?			2.0E-3	7.5E-1	7.5E-1
SD-XHE-XM-FINJ	Operator Fails RCS Injection using FLEX SPC (4306.01P004 Section 4.1)	4306.01P004 Section 4.1	1 Hour		2E-3	2E-3	4E-3
FC-XHE-XM-MCV	Manually Venting of Containment with 1FC012A & B	CPS 4303.01P001		>24 Hours	4E-3	2E-4	4.2E-3
SD-XHE-XM-ISDC	Isolate SDC after LOSDC	4006.01					

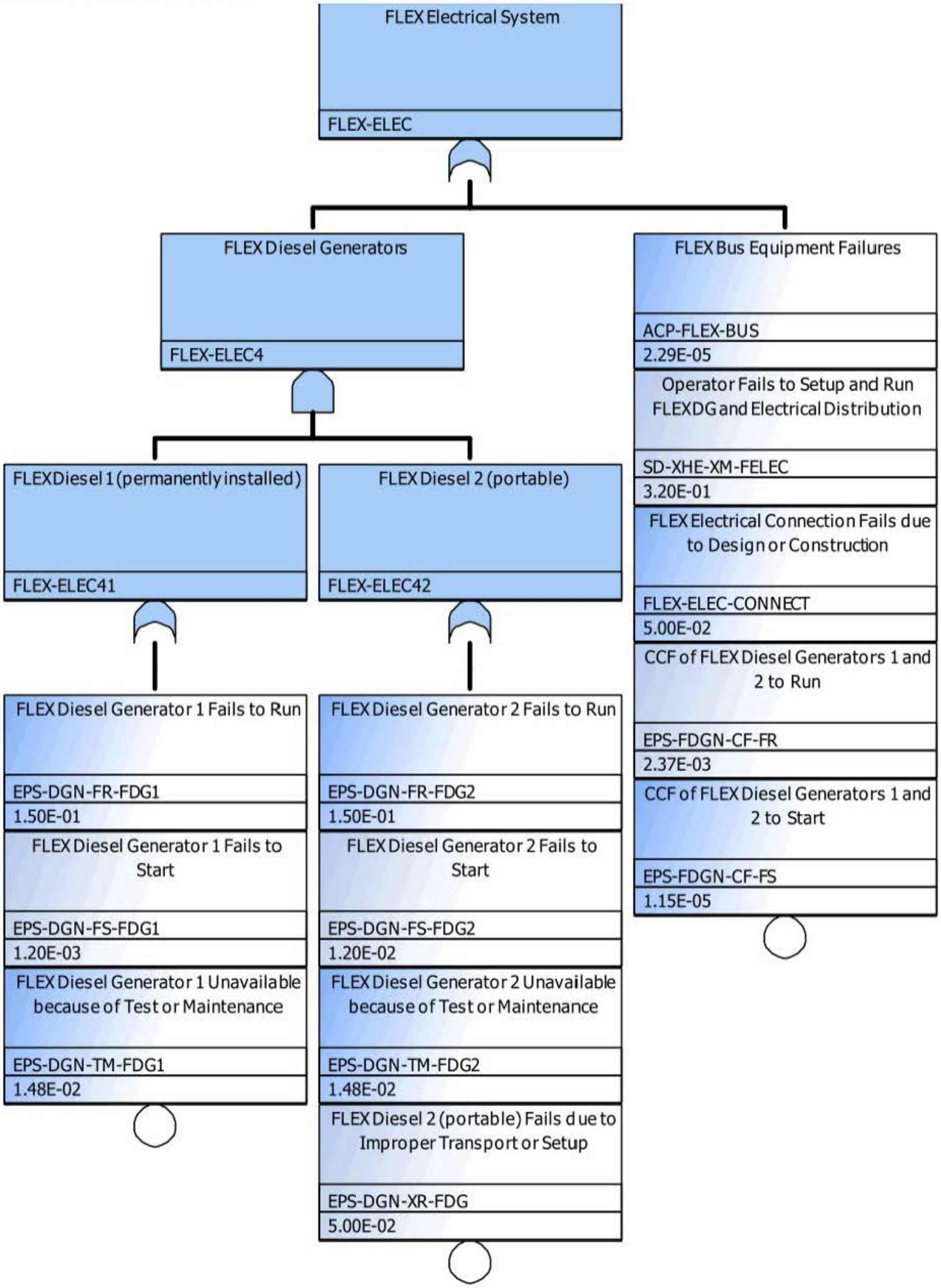
Div.1 DC Bus with Credit for FLEX Fault Tree



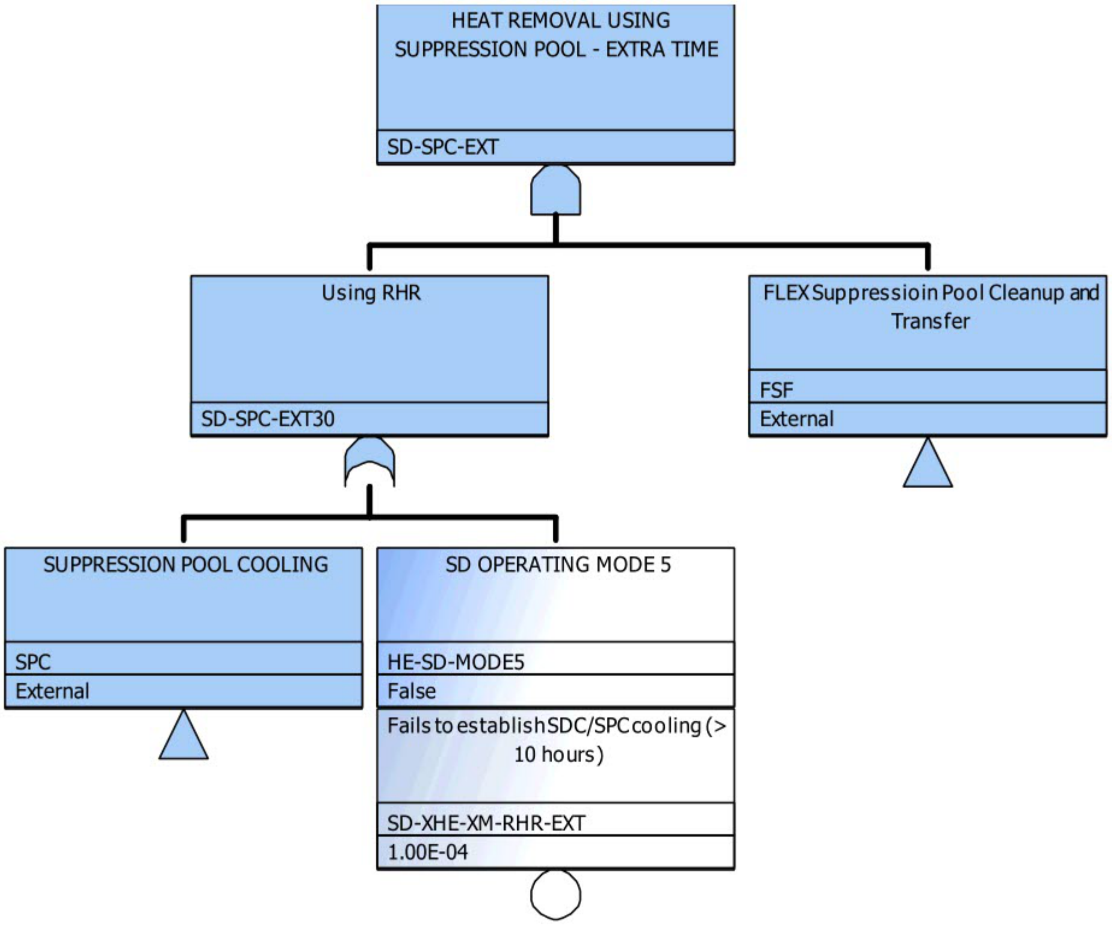
Div. II DC Bus with Credit for FLEX Fault Tree



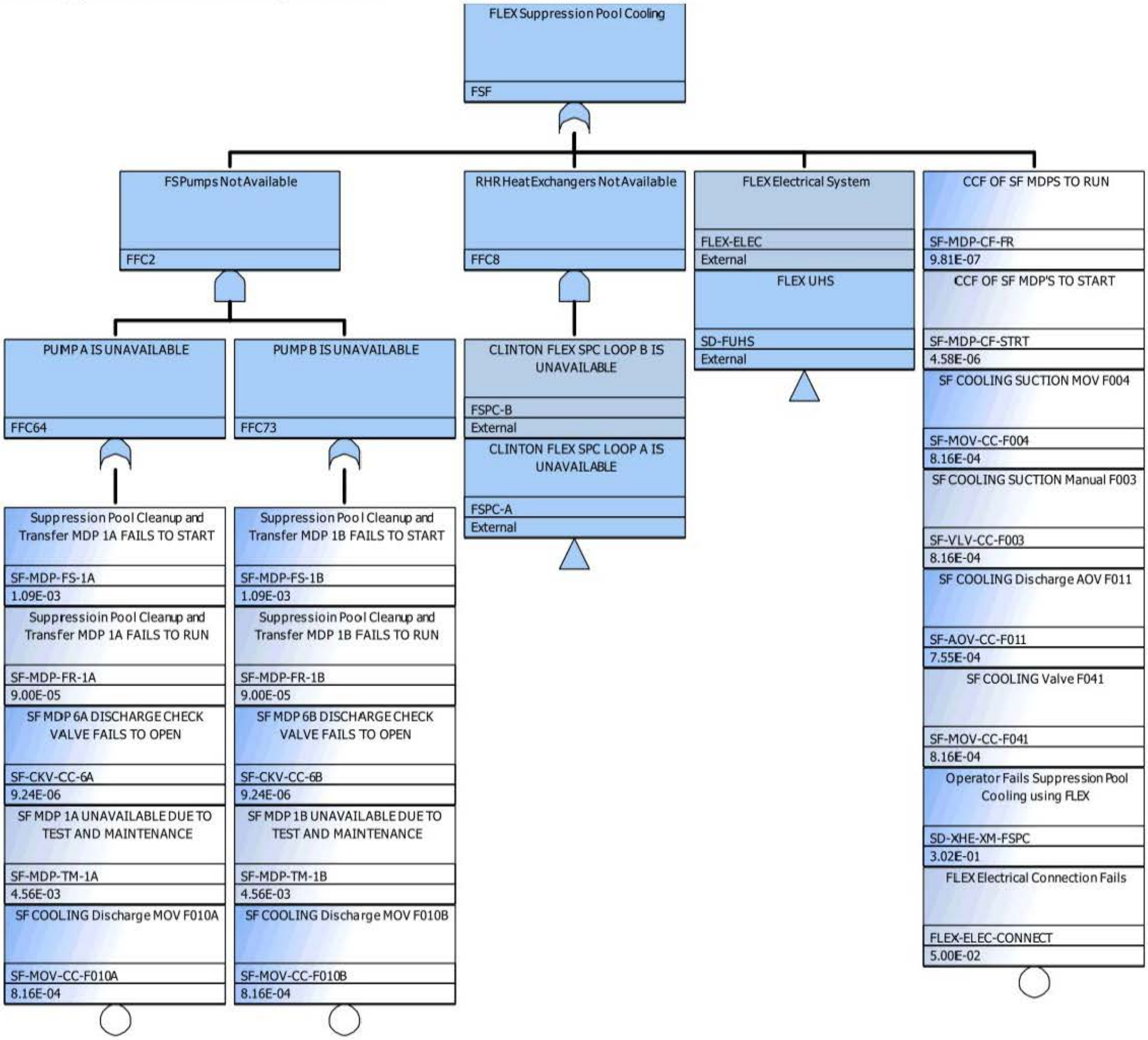
FLEX AC Electrical Fault Tree



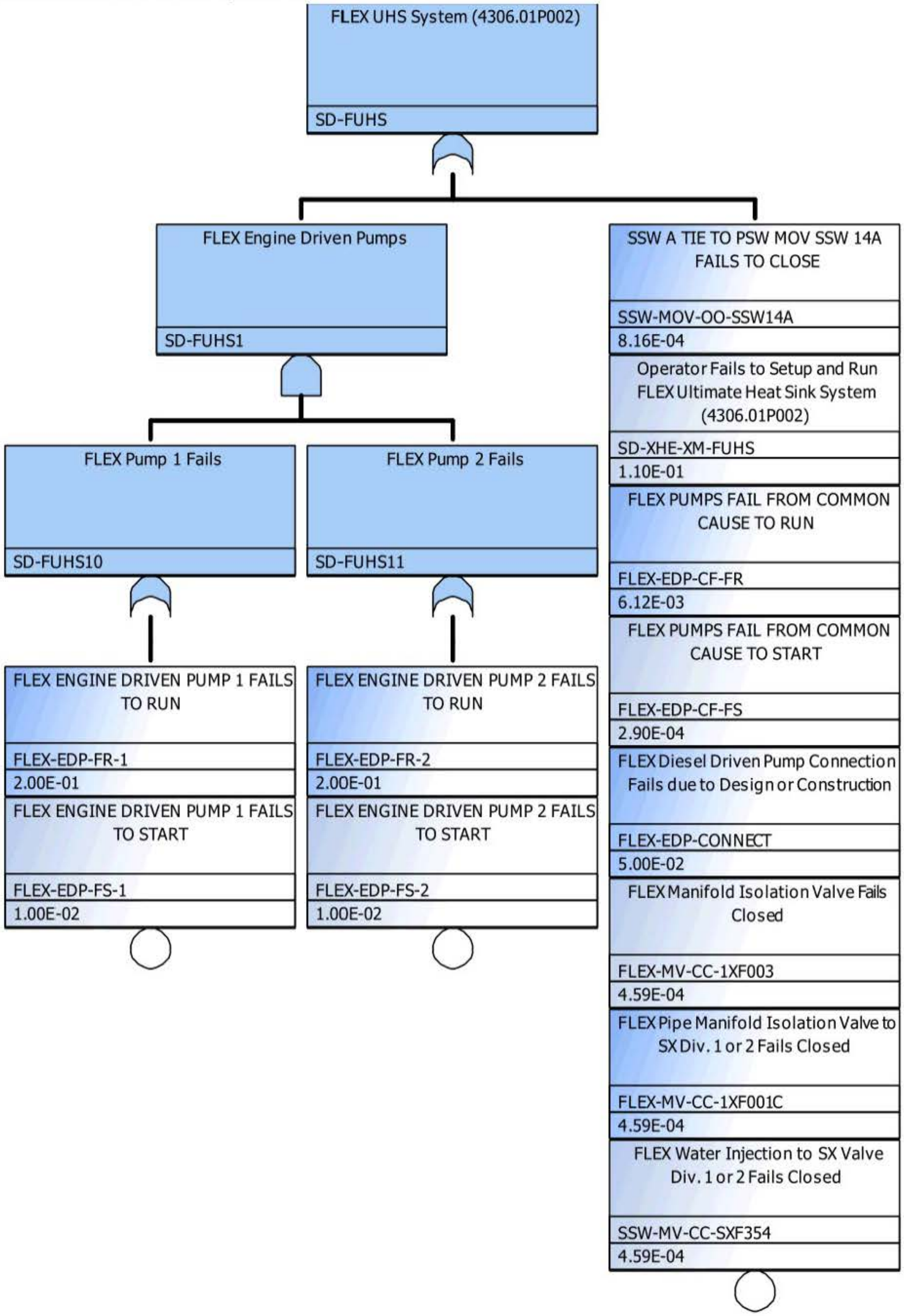
Suppression Pool Cooling Crediting FLEX Fault Tree



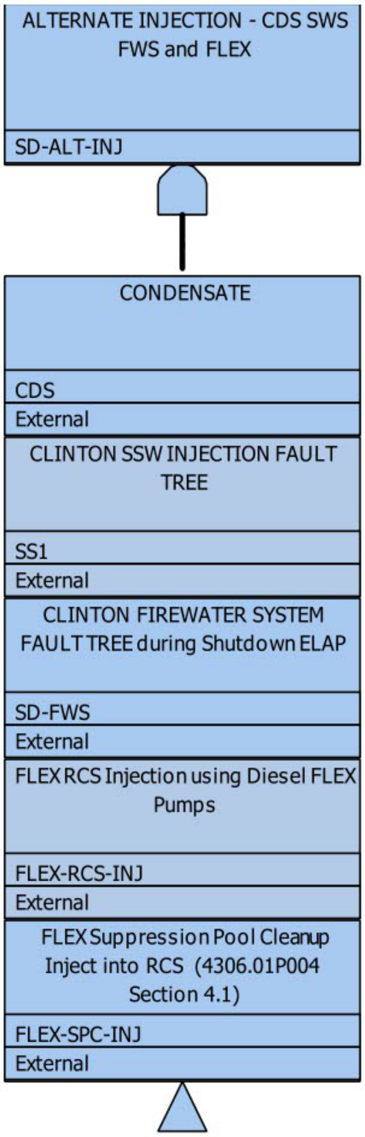
FLEX Suppression Pool Cooling Fault Tree



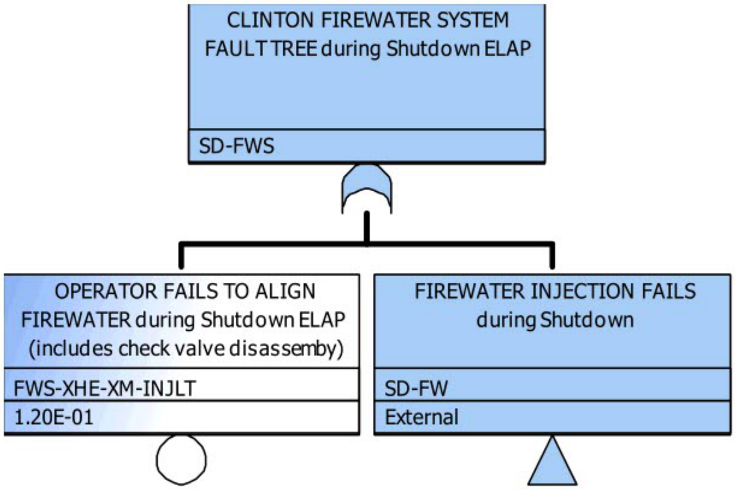
FLEX Ultimate Heat Sink System Fault Tree



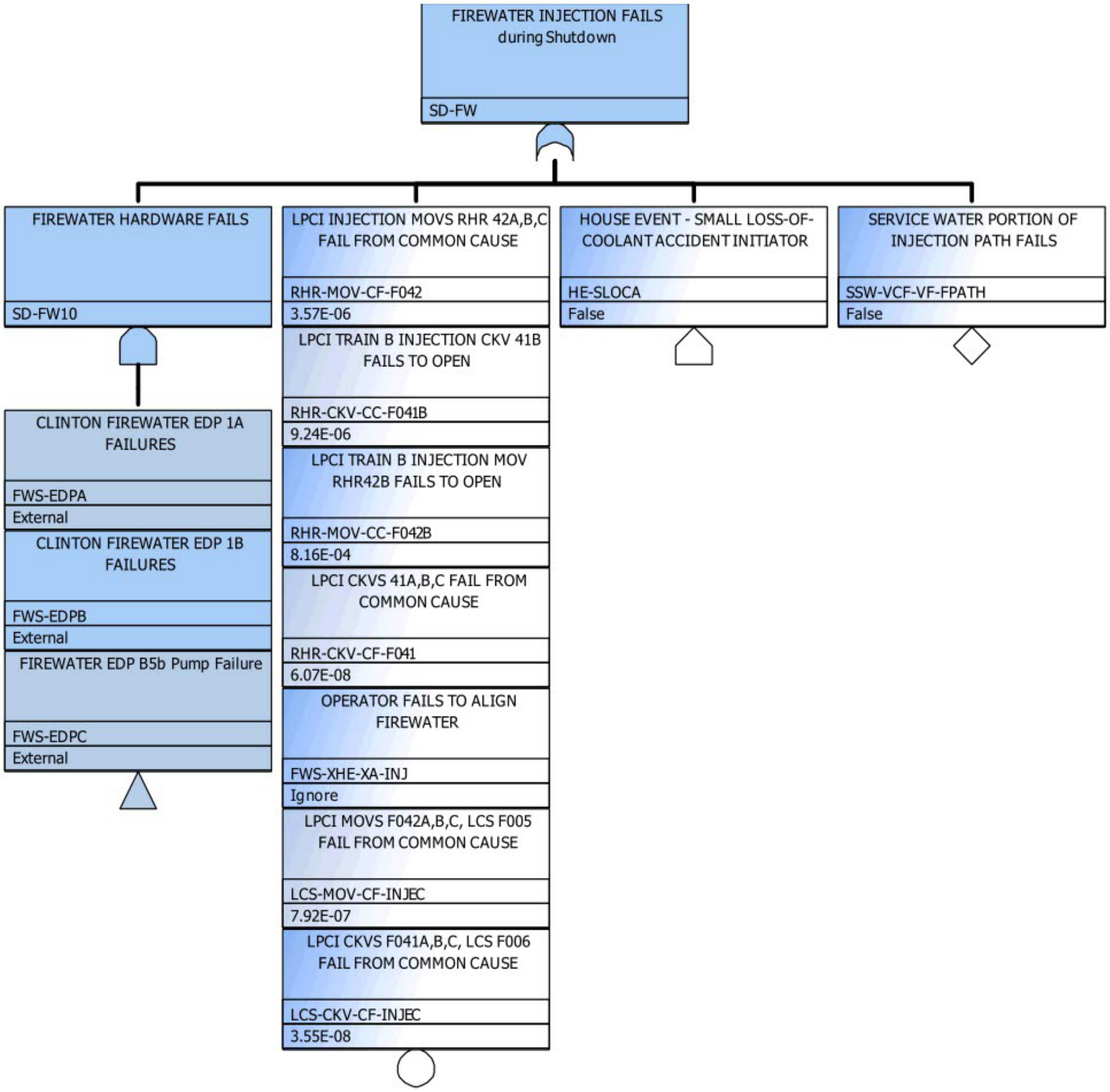
Alternate Injection CDS, SWS, SWS and FLEX Fault Tree



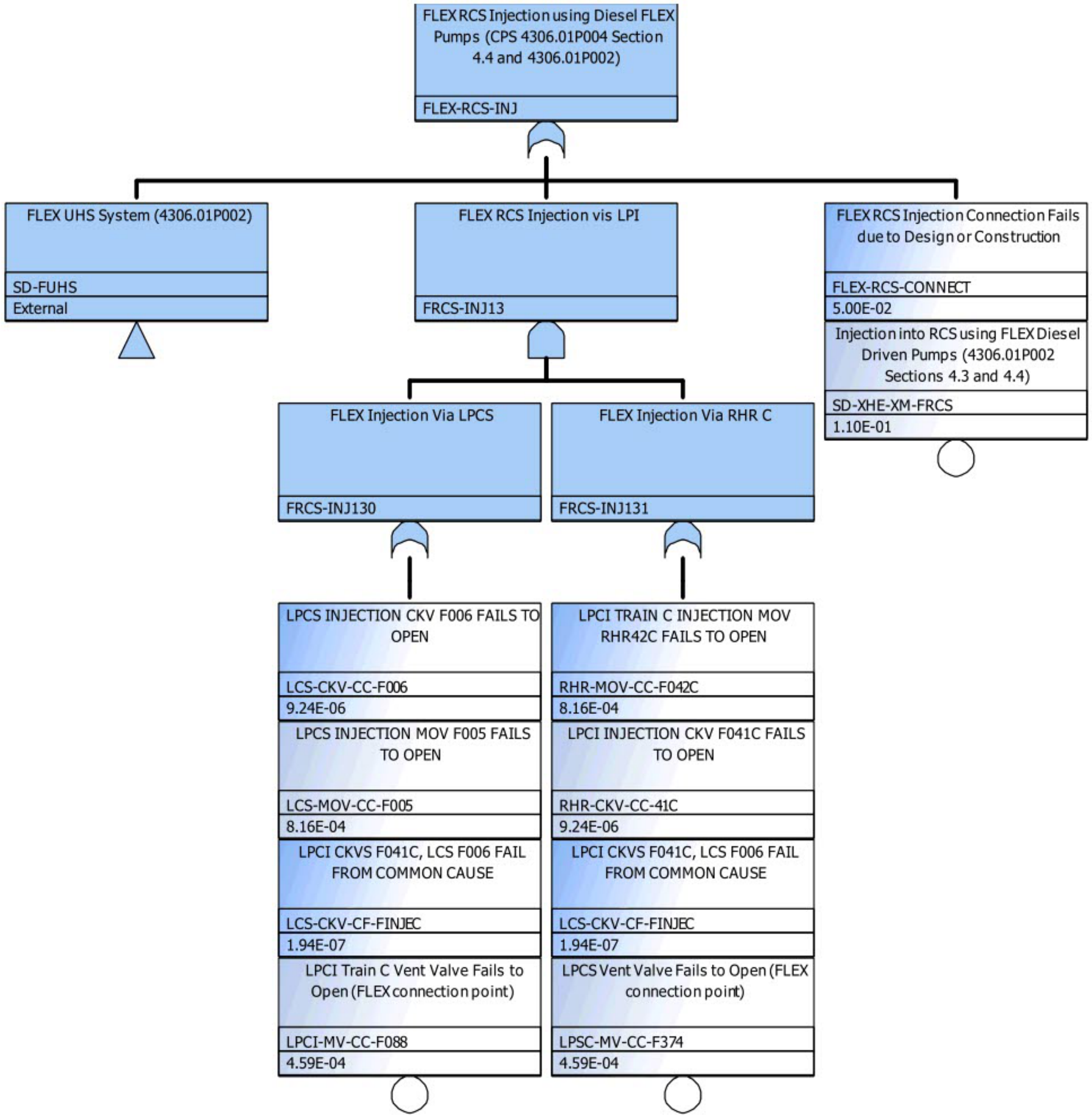
Firewater Injection into RCS Fault Tree



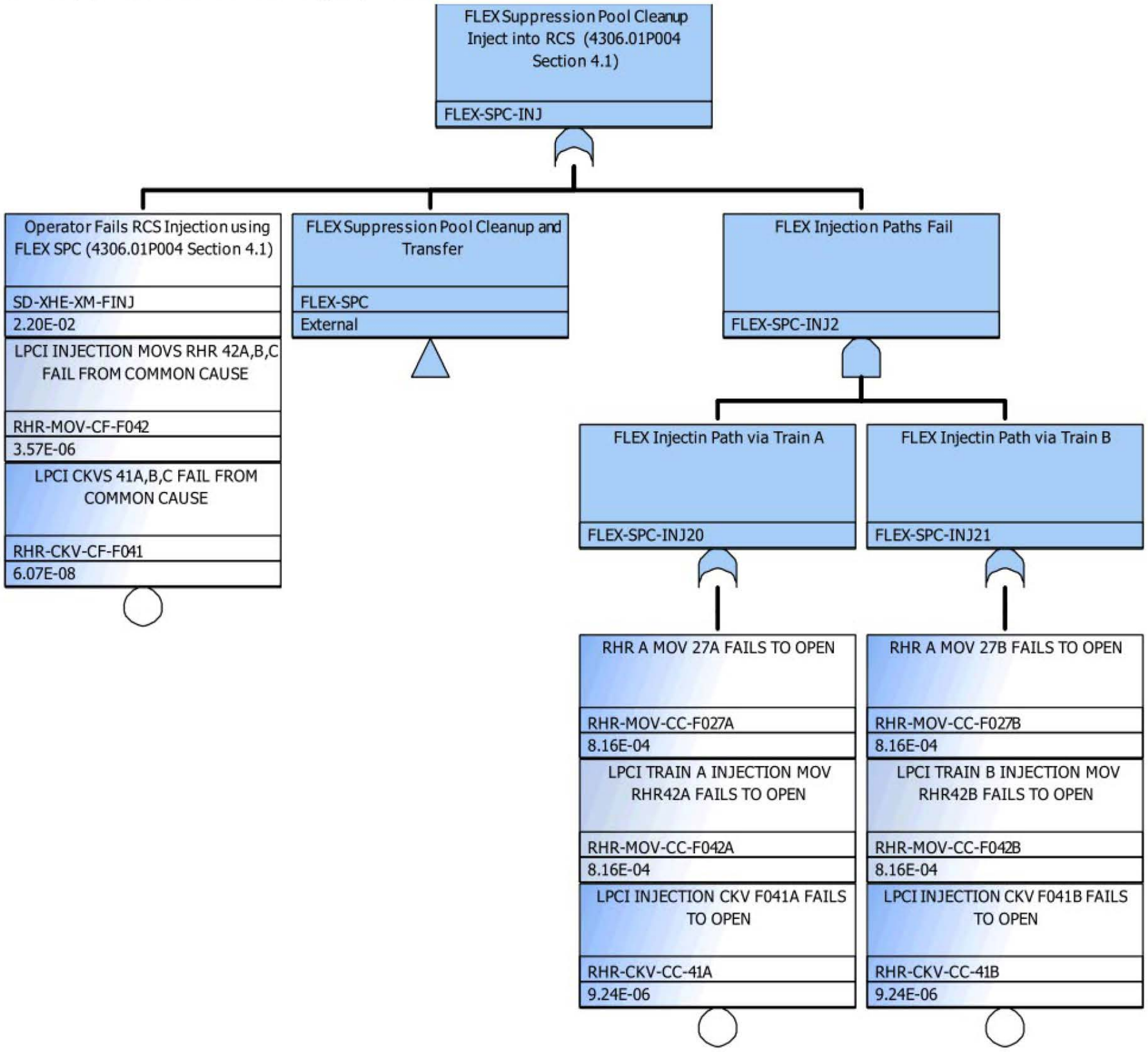
Firewater Equipment Fault Tree



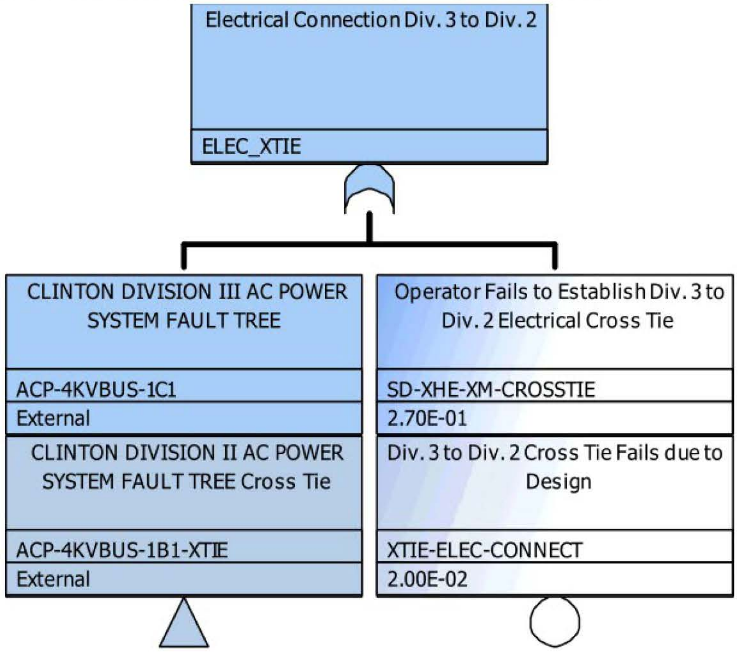
FLEX RCS Injection Using Diesel Driven Pumps Fault Tree



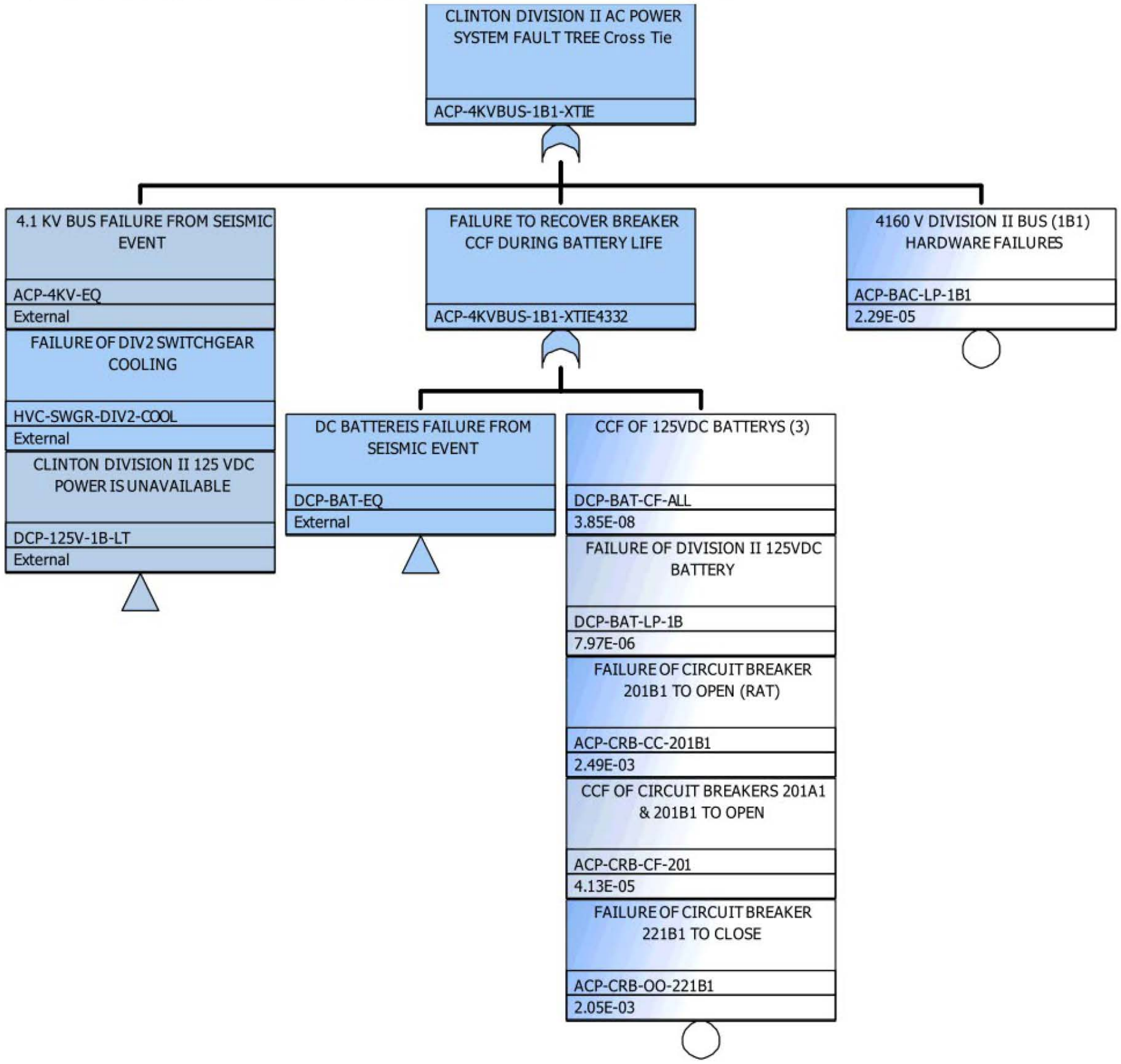
FLEX Suppression Pool Cooling Injection to RCS Fault Tree



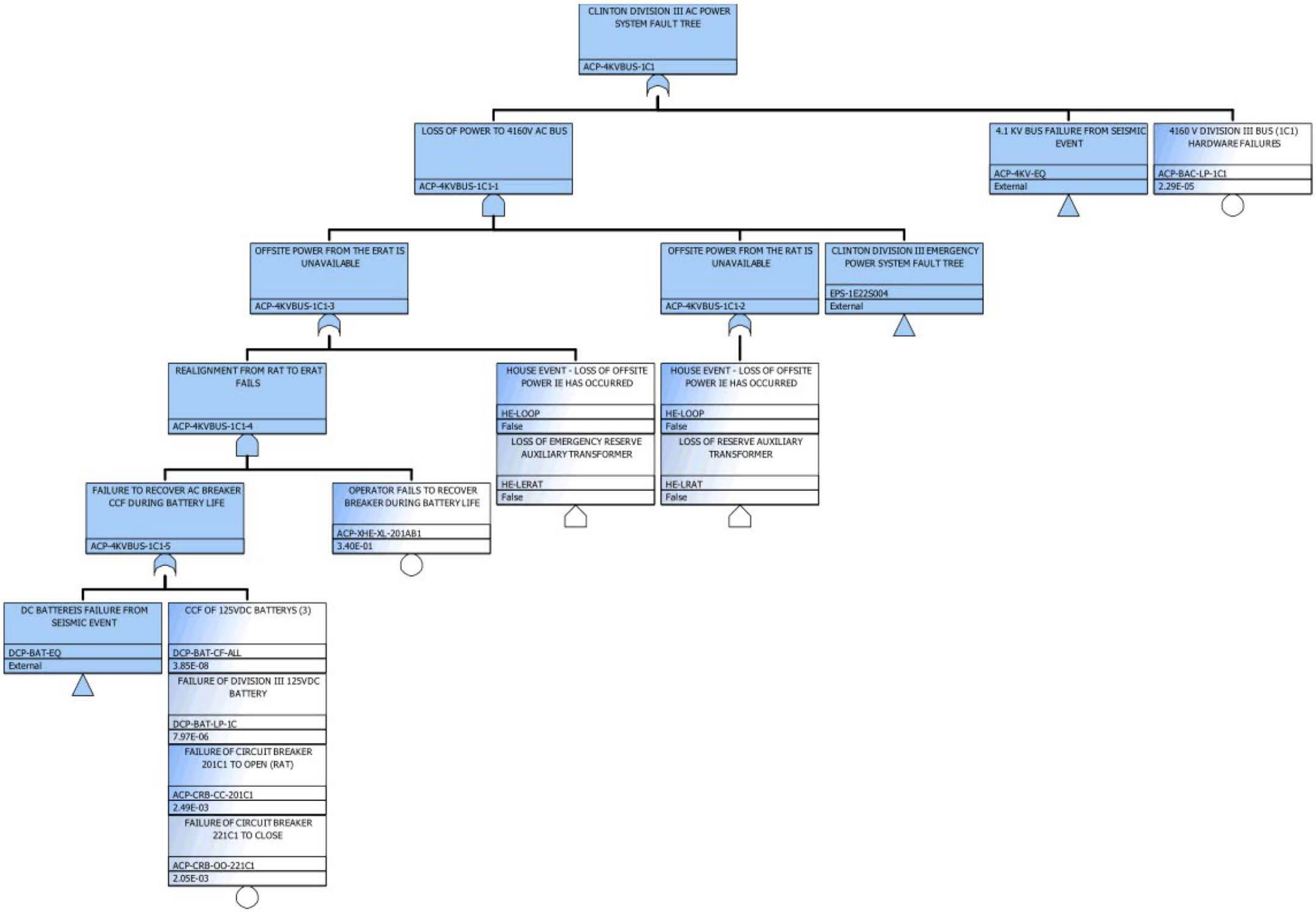
Electrical Cross Tie between Div. 3 and Div. 2 Fault Tree



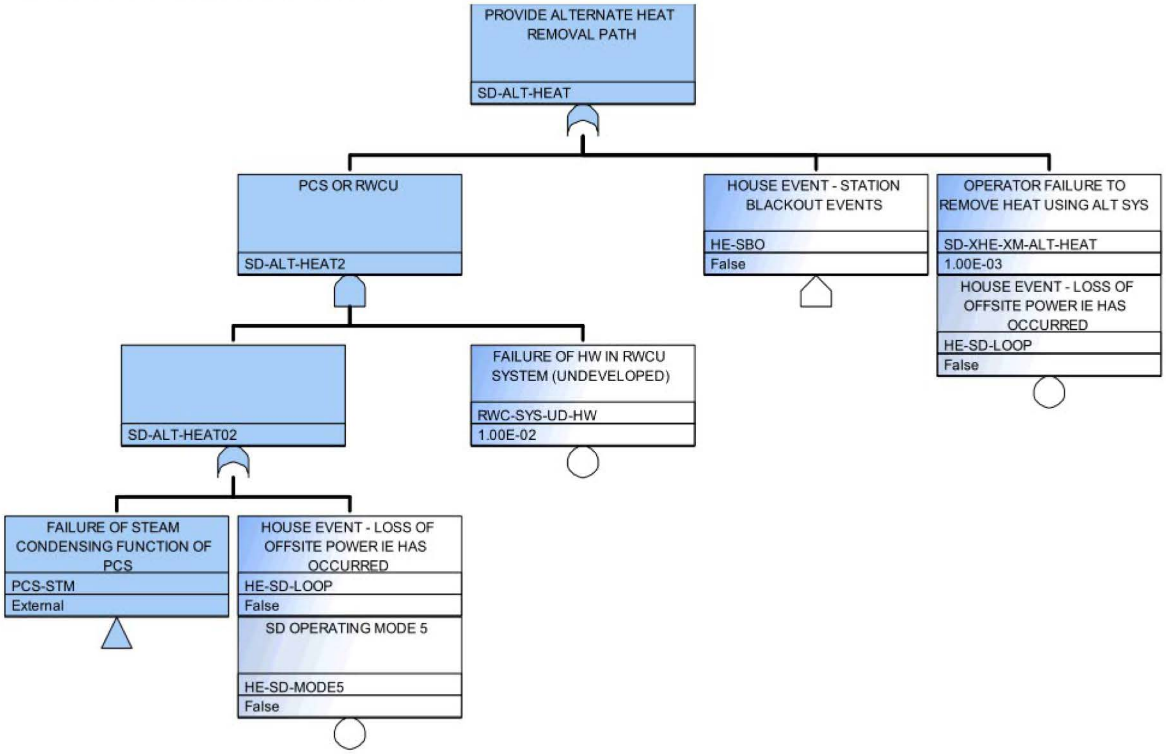
Division II AC Power Electrical for Cross Tie to Division III Fault Tree



Division III AC Power Fault Tree



Alternate Heat Removal Fault Tree



Clinton re-SERP Pre-briefing Questions

02-17-2019

J. Mitman

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Clinton Shutdown SPAR model

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Summary from Special Inspection Report

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PSFs that are assumed to be other than Nominal

HEP	**PSF Setting**
SD-XHE-XM-FRCS – Fail RPV inject with FLEX pumps	Obvious Diagnosis Complexity (Diagnosis) Moderate Complexity (Action) Low Experience/Training (Action) Poor Ergonomics (Action)

PSFs assumed correctly. No errors or omissions in report. PSFs not described because this is the less preferred FLEX injection method and is not currently important to the overall results.

SD-XHE-XM-DCLS – Fail to load shed	Moderate Complexity (Diagnosis) Poor Ergonomics (Action)

PSFs assumed correctly. No errors or omissions in report. No assumption on HEP documented because this HEP is not currently important to the overall results.

SD-XHE-XM-FRCIC – Fail to operate RCIC	Obvious Diagnosis Complexity (Diagnosis)

PSFs assumed correctly. No errors or omissions in report. Assumption states that HEP is dominated by action.

FC-XHE-XM-MCV – Fail Manual containment venting	Extra Time (Diagnosis) >=5x Time (Action) Moderate Complexity (Diagnosis)

PSFs assumed correctly. No errors or omissions in report. PSFs not described because this HEP is not currently important to the overall results.

SD-XHE-XL-ELAP – Fail to recover Electrical	Moderate Complexity (Action)

PSFs assumed correctly. Omission in description of PSFs. Will change report to add moderate complexity. Values of the HEP correct in assumptions, table 2 and the fault tree.

Observations on Division 3 to Division 1 or Division 2 cross-tie

- Action does not appear to be a reliable means to recover decay heat removal in scenario of interest (SBO, HPCS providing injection/inventory control, no decay heat removal)

- HPCS must be stopped to align the cross-tie, putting the plant in a condition with no injection.

- HRA analysis estimates 2 to 4 hours to install cross-tie based on a walkdown. Allowable time not documented in HRA analysis. Based on SRA question, licensee estimates 4 hours available based on MAAP evaluation.

- HRA analysis focused on providing power to Division 1 for operation of an RHR and SX pump. Division 3 DG is not large enough to power HPCS, RHR, and SX at the same time, requiring operators to manage loads (doesn't appear to be evaluated in HRA, is practiced in simulator?)

- Licensee statement that many of the actions can be achieved on the receiving bus with no impact on HPCS. This conflicts with the procedure that states "Steps in this procedure must be performed in order to avoid the risk of personnel injury or equipment damage."

- Licensee statement that this procedure could be used to power containment vent paths. HRA analysis has the following statement: "Based on crew interview, operators question the value of this action. Use of this procedure could result in a loss of HPCS injection. Containment venting is not available without Division 1 and Division 2 AC power. The utility has a policy of verbatim compliance to procedures." The procedure provides instructions to power either Division 1 or Division 2. If both divisions are needed for containment venting then this procedure will not support containment venting for decay heat removal.

- Training appears to be very limited. The licensee provided a presentation that provides a reference to the procedure, shows the line-up to be achieved, with an overview (not detailed) of the actions necessary. Is this the training that is provided every two years?

- Procedure was complex. Reference to Figure 3, step 1.2 did not appear the same in the field.

Note to requester: The document on this page and the next page are notes from a call between NRR and Clinton on October 15, 2018 that are being withheld in their entiety under FOIA Excemption B5 (deliberative process priviledge).

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Clinton Power Station White Finding for Multiple Emergency Diesel Inoperability

Key Messages

- **On April 1, 2019, the NRC issued NRC Inspection Report No. 05000461/2018092, (ML19092A212) documenting a final significance determination of White for failing to accomplish activities affecting quality in accordance with multiple procedures and consequently operating the plant in violation of Title 10 CFR Part 50, Appendix B, Criterion V, "Instructions, Procedures, and Drawings," and the plant's Technical Specifications. This finding moves Clinton to Column 2 of the Action Matrix.**

- **The NRC took into consideration: information developed during a Special Inspection, the information provided at the regulatory conference, and the supplemental information that Clinton provided.**

- **The NRC noted that this was a complex shutdown significance determination. NRC Inspection Report No. 05000461/2018092 documents a number of areas of disagreement between the NRC analysis and licensee assertions and provides rational for the NRC conclusions.**

Facts

- As a result of several human performance errors the station's Division 2 Emergency Diesel Generator (EDG) was inoperable and unavailable for over 6 days without the licensee's knowledge. Both Division 1 and Division 2 EDGs were inoperable and unavailable for over 3 days, May 14 through May 17, 2018, which was not allowed per Technical Specification (TS) 3.8.2. NRC Special Inspection Report 05000461/2018050 dated August 23, 2018 (ML18235A170) documents this apparent violation and that the condition was a result of the licensee failing to follow multiple equipment configuration control procedures. NRC Inspection Report 05000461/2018051 dated October 15, 2018, (ML18289A436) documents the preliminary White significance of the finding.

- Clinton Power Station has been in Column 2 of the Action Matrix since July 1, 2017. A White finding associated with the Division 1 EDG room ventilation fan was identified in the 3rd quarter of 2017 and cleared by IP95001 inspection in the 2nd quarter 2018. A White finding associated with a Division 3 shutdown service water (SX) pump failure was identified in the 4th quarter of 2017 and cleared by IP95001 inspection in the 4th quarter of 2018. The start date of the current White finding is assigned to 3rd quarter of 2018 in accordance with the date of the Special Inspection Report.

3Q 2017	4Q 2017	1Q 2018	2Q 2018	3Q 2018	4Q 2018	1Q 2019	2Q 2019
White 1	White 1	White 1	White 1				
	White 2	White 2	White 2	White 2	White 2		
				White 3	White 3	White 3	White 3

- Clinton Power Station's End of Cycle Assessment letter noted a return to column 1 on November 28, 2018 with a caveat that there was a preliminary White finding in process.

- Subsequent to the final significance determination of White, a FOIA request was submitted to the agency, by a lawyer who attended the November 2018 Regulatory Conference, requesting all information related to this finding.

Proposed/Possible New AC Power Recovery FT and ELAP NRP

(b)(5)

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1. Are there any other airlocks besides those at 737 and 828?

2. What was the status of the primary containment airlocks during the entire time of the unavailability of the Div. 2 EDG?

3. Where there any other openings in containment that would prevent containment pressurization upon loss of SDC and subsequent steaming into primary containment during the entire period of Div. 2 EDG unavailability?

4. Exelon supplied a containment closure procedure. If penetrations were open, is there any reason why primary containment could not or would not be closed during any period of Div. 2 EDG unavailability?

5. Base HEP analysis assumes appropriate instrumentation is available. Of most importance is RPV level. The two ranges that appear to be calibrated for the conditions of interest are the shutdown and fuel zone instruments. What level instruments were available during the period of Div. 2 EDG unavailability?

6. Please provide reliability data for FLEX equipment (number of starts, duration of runs, and number of failures to start or run). Please provide failure probabilities derived from the previous date. For equipment failures, please provide the associated condition report.

7. For the simulator exercise related to FLEX, was the Division 3 DG available? The guide indicates the HPCS pump shaft was damaged but is silent on the availability of the DG.

8. Training material for the Division 3 cross-tie includes a 5 minute discussion of the procedure during an EDMG-based simulator exercise and a DBIG training session covering all EDMGs with 2 slides showing the cross-tie procedure and a diagram of the electrical line-up. Does training include required walk-throughs of the procedure in the field?

Risk Insights from NRC SDP on Clinton EDG Issue

March 2019

1. Exelon did not have spare electrical breakers for the FLEX riser. When asked about this during the NRC's site visit (October 2018), they order several.
2. During loses of shutdown cooling, maintaining the reactor coolant system (RCS) at low pressure increases the time to core uncovery in contrast to allowing the RCS to pressurize to normal operating temperature and pressure. If the intent of
3. SDC isolation
4. Declaring ELAP

From: Leech, Matthew
Sent: Mon, 5 Nov 2018 17:46:27 +0000
To: Mitman, Jeffrey
Subject: Clinton HFE
Attachments: Matt's HFE Task Analysis - SD- EPS-XHE-XM-NR10H (EDG2) Re-evaluation.doc

My comments are on page 7.

Matt Leech
US NRC
Reliability and Risk Analyst
(301) 415- 8312

Human Failure Event (HFE) ID: SD-EPS-XHE-XM-NR01H

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From: Miller, Chris
Sent: Thu, 28 Feb 2019 14:07:22 +0000
To: Lara, Julio
Subject: Clinton SERP

Hi Julio,

(b)(5)

thanks
chris

Christopher Miller
Director, Division of Inspection and Regional Support
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
301-415-1004

From: Miller, Chris
Sent: Mon, 25 Mar 2019 14:20:11 +0000
To: Nieh, Ho;Felts, Russell;Evans, Michele;McDermott, Brian
Subject: FW: Clinton White

We in DIRS and DRA concurred on the Detailed Risk Evaluation changes. The concurrence on the letter is upcoming.
chris

From: Lara, Julio
Sent: Monday, March 25, 2019 9:01 AM
To: Miller, Chris <Chris.Miller@nrc.gov>; Dickson, Billy <Billy.Dickson@nrc.gov>
Subject: RE: Clinton

It is in concurrence. It should be in your "in-box" today or tomorrow. As of Friday, OE/DRA/DIRS were next.

From: Miller, Chris
Sent: Monday, March 25, 2019 7:58 AM
To: Lara, Julio <Julio.Lara@nrc.gov>; Dickson, Billy <Billy.Dickson@nrc.gov>
Subject: Clinton

Has the letter with White significance gone out to licensee yet?
Thanks
Chris

From: Mitman, Jeffrey
Sent: Fri, 16 Nov 2018 21:28:36 +0000
To: Fong, CJ
Cc: Kozak, Laura;Kichline, Michelle;Montecalvo, Michael;Leech, Matthew
Subject: HFE Task Analysis - SD- EPS-XHE-XM-NR10H (EDG2) Post SERP Rev.0.doc
Attachments: HFE Task Analysis - SD- EPS-XHE-XM-NR10H (EDG2) Post SERP Rev.0.doc

CJ,

Attached is the revised HFE analysis of recovery of EDG2 that Mike Franovich requested. The only change to this write-up is the inclusion of three additional data points representing the reanalysis by the three DRA/APHB risk analysts with operational backgrounds. [redacted] (b)(5)
(b)(5) [redacted]

(b)(5) [redacted]

Thanks to Michelle, Matt and Mike for their help.

Jeff Mitman

Human Failure Event (HFE) ID: SD-EPS-XHE-XM-NR01H

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From: Kichline, Michelle
Sent: Thu, 25 Oct 2018 18:39:30 +0000
To: Mitman, Jeffrey
Subject: RE: Re-evaluation of HFE for EDG non recovery - Clinton

I have some questions:

- When the EDG fails to start, how many alarms related to the EDG failure will annunciate in the MCR? Is there a general EDG trouble alarm? Where do the MCR alarms point the operator?
- When are the crews trained to go to ELAP? The definition of ELAP presented in the LOAC procedure very clearly states that there needs to be a BDBEE to declare an ELAP, so I don't see why a LOOP would result in a declaration of an ELAP without some severe external stuff going on. However, the LOAC procedure doesn't get any more helpful if they don't declare an ELAP because they are still going to load shed.

My evaluation, based on 1 hour to diagnose the reason for the EDG failure and correct it:

Relevant <u>Diagnostic</u> Performance Shaping Factors (PSF)

- Time: Extra time (.1)
- Stress: high (2)
- Complexity: highly complex (5)
- Experience/Training: nominal (1)
- Procedures: not available (50)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .5 (I think it could be higher, like .7, but SPAR-H goes over 1 then. I think complexity should have more of an affect, and procedures should have less of an impact. If there were less than an hour available, I would estimate a 1.0.)

Relevant <u>Action</u> Performance Shaping Factors

- Time: nominal (1)
- Stress: nominal (1)
- Complexity: nominal (1)
- Experience/Training: nominal (1)
- Procedures: nominal (1)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .001 (the action part of this HEP is simple and does not require a procedure)

HEP Calculation

Diagnosis: .5
Action: .001

Final HEP (Independent)	**5.E-01**
Final HEP (Low Dependence)	**5.E-01**
Final HEP (Moderate Dependence)	**6.E-01**
Final HEP (High Dependence)	**8.E-01**

Final HEP (Complete Dependence)	1

Michelle Kichline
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3153

From: Mitman, Jeffrey
Sent: Monday, October 22, 2018 11:54 AM
To: Montecalvo, Michael <Michael.Montecalvo@nrc.gov>; Kichline, Michelle <Michelle.Kichline@nrc.gov>; Leech, Matthew <Matthew.Leech@nrc.gov>; Demers, Jerrod <Jerrod.Demers@nrc.gov>; Hartle, Brandon <Brandon.Hartle@nrc.gov>
Cc: Fong, CJ <CJ.Fong@nrc.gov>
Subject: Re-evaluation of HFE for EDG non recovery - Clinton

Mike Franovich has requested that I poll the branch for their insights and input into a significant HFE on the Clinton SDP. The SDP address a 3.5 day period during their most recent refueling outage during which neither the Div. 1 nor 2 EDG was available. The SERP has determined that the finding is preliminarily White. A choice letter has been written and the licensee has requested a Reg. Conf.

The HFE looks at the non-recovery probability of the inadvertently unavailable EDG2. It is a dominant HFE and is driving the results.

The purpose of the re-evaluation is to use it as additional sensitivity analysis and input into the final decision making.

Attached is the HFE analysis itself minus the quantification. I have not supplied the quantification as I don't want it to influence your analysis. Also attached are the annunciator response card for the associated annunciators. Finally, attached are the relevant procedures.

Hopefully, the HFE analysis document will supply all of the information needed to understand the scenario and what the operators would have faced. In reality, you will probably have questions.

Please review the HFE document. Also the procedures to the degree you feel necessary. I'll try to find a time slot after the SRA counterparts meeting this week to meet as a group to answer any questions and to go through the quantification.

There is one additional piece of information that I want everyone to have. The non-recovery probability for an EDG based on data is 0.88 for the one hour available.

Thanks for the help.

Jeff Mitman

All but the first attachment is non-responsive due to narrowing the request to exclude licensee orginated documents. Documents excluded is are: 5285_R27c Alarm Panel 5285 Annunciators; 3506.01_R38 EDG and Support Systems; CPS 4200.01 Loss of AC Power; and CPS 3501.01 High Voltage Auxiliary Power System.

From: Mitman, Jeffrey
Sent: Mon, 22 Oct 2018 15:54:13 +0000
To: Montecalvo, Michael;Kichline, Michelle;Leech, Matthew;Demers, Jerrod;Hartle, Brandon
Cc: Fong, CJ
Subject: Re-evaluation of HFE for EDG non recovery - Clinton
Attachments: HFE Task Analysis - SD- EPS-XHE-XM-NR10H (EDG2) Re-evaluation.doc, 5285_R27c ALARM PANEL 5285 ANNUNCIATORS AT 1PL12JB.pdf, 3506.01_R38 EDG and Support Systems.pdf, 4200.01, Loss of AC Power.pdf, CPS 3501.01 High Voltage Auxiliary Power System.pdf

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Thanks for the help.

Jeff Mitman

Human Failure Event (HFE) ID: SD-EPS-XHE-XM-NR01H

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From: Kichline, Michelle
Sent: Mon, 29 Oct 2018 18:25:56 +0000
To: Mitman, Jeffrey
Subject: Re-evaluation of my Re-evaluation of HFE for EDG non recovery - Clinton

My previous evaluation assumed that the only way to find the problem was for someone to walk in the room and notice the valves were out of place because it seemed likely to me that DC load shedding would occur quickly, before troubleshooting could occur or the lineup procedures could be referenced. Based on our discussion, I have re-evaluated the diagnosis portion of the HEP assuming that there is a full hour available to perform the action before any load shedding or ELAP declaration occurs. Therefore, licensee staff will have time to go through the EDG operating procedure to get to the step that will instruct them to close the valves. I also used your excel spreadsheet, which appears to be calculating the adjusted HEP correctly. My spreadsheet was not. That makes the diagnosis HEP .168 when adjusted for the 3 negative PSFs. There are no changes to the action part of the evaluation.

Relevant Diagnostic Performance Shaping Factors

- Time: Nominal (1)
- Stress: high (2)
- Complexity: moderately complex (2)
- Experience/Training: nominal (1)
- Procedures: available but poor (5)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .2 without adjustment, .168 with adjustment for negative PSFs.

Relevant Action Performance Shaping Factors

- Time: nominal (1)
- Stress: nominal (1)
- Complexity: nominal (1)
- Experience/Training: nominal (1)
- Procedures: nominal (1)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .001 (the action part of this HEP is simple and does not require a procedure)

From: Kichline, Michelle
Sent: Thursday, October 25, 2018 2:40 PM
To: Mitman, Jeffrey <Jeffrey.Mitman@nrc.gov>
Subject: RE: Re-evaluation of HFE for EDG non recovery - Clinton

Relevant Diagnostic Performance Shaping Factors (PSF)

- Time: Extra time (.1)
- Stress: high (2)
- Complexity: highly complex (5)
- Experience/Training: nominal (1)

- Procedures: not available (50)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .5 without adjustment, .336 with adjustment for 3 negative PSFs (I think it could be higher, like .7, but SPAR-H goes over 1 then. I think complexity should have more of an affect, and procedures should have less of an impact. If there were less than an hour available, I would estimate a 1.0.)

Relevant Action Performance Shaping Factors

- Time: nominal (1)
- Stress: nominal (1)
- Complexity: nominal (1)
- Experience/Training: nominal (1)
- Procedures: nominal (1)
- Ergonomics/FFD/Work Processes: Nominal (1)

Total: .001 (the action part of this HEP is simple and does not require a procedure)

HEP Calculation

Diagnosis: .5

Action: .001

Final HEP (Independent)	**5.E-01**
Final HEP (Low Dependence)	**5.E-01**
Final HEP (Moderate Dependence)	**6.E-01**
Final HEP (High Dependence)	**8.E-01**
Final HEP (Complete Dependence)	**1**

Michelle Kichline
Division of Risk Assessment
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
301-415-3153

From: Mitman, Jeffrey
Sent: Monday, October 22, 2018 11:54 AM
To: Montecalvo, Michael <Michael.Montecalvo@nrc.gov>; Kichline, Michelle <Michelle.Kichline@nrc.gov>; Leech, Matthew <Matthew.Leech@nrc.gov>; Demers, Jerrod <Jerrod.Demers@nrc.gov>; Hartle, Brandon <Brandon.Hartle@nrc.gov>
Cc: Fong, CJ <CJ.Fong@nrc.gov>
Subject: Re-evaluation of HFE for EDG non recovery - Clinton

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Hopefully, the HFE analysis document will supply all of the information needed to understand the scenario and what the operators would have faced. In reality, you will probably have questions.

Please review the HFE document. Also the procedures to the degree you feel necessary. I'll try to find a time slot after the SRA counterparts meeting this week to meet as a group to answer any questions and to go through the quantification.

There is one additional piece of information that I want everyone to have. The non-recovery probability for an EDG based on data is 0.88 for the one hour available.

Thanks for the help.

Jeff Mitman

From: Bakhsh, Sarah
To: Bakhsh, Sarah; Bickett, Brice; Bigoness, Jay; Bollock, Douglas; Burgess, Michele; Carpenter, Robert; Casey, Lauren; Chyu, Doris; Crisden, Cherie; Figueroa Toledo, Gladys; Fretz, Robert; Furst, David; Gibbs, Russell; Gulla, Gerald; Harrison, John; Hasan, Nasreen; Hilton, Nick; Holiday, Sophie; Hollcraft, Zachary; Jayroe, Peter; Justice, Jared; Kowal, Mark; Kozak, Laura; Kramer, John; Lambert, Kenneth; Marenchin, Thomas; McLaughlin, Marjorie; NRREnforcement Resource; Peduzzi, Francis; Purdy, Gary; Richardson, Alonzo; Simonian, Niry; Solorio, Dave; Sparks, Scott; Sreenivas, Leelavathi; Sun, Robert; Vrahoretis, Susan; Warnek, Nicole; White, Duane; Woods, Susanne; Jones, David; Marshfield, Mark; Lemoncelli, Mauri; Peralta, Juan; Hanna, John; Garmoe, Alex; Ruesch, Eric; Hawkins, Sarenee; Kent, Jonathan; Vasquez, Michael; Franklin, Carmen; Bigoness, Jay; Aird, David; Rajapakse, Champa; Torres, Edgardo; Wilson, George; Solomakos, Matina
Subject: SENSITIVE PRE-DECISIONAL INFORMATION: Enforcement Highlights April 2, 2019
Date: Tuesday, April 02, 2019 10:47:34 AM
Attachments: Enforcement Highlights April 2, 2019.pdf

Good morning,

The attached table contains pre-decisional enforcement information. Please do not distribute without prior EICS approval.

Please contact Sarah Bakhsh [630 810 4380], Kenneth Lambert [630 810 4376], or Paul Pelke [630 810 4375] if you have questions.

U.S. NUCLEAR REGULATORY COMMISSION

- - - WARNING - - -

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NRC FORM 762
(9-2005)

Allegation and Enforcement Status

April 2, 2019

~~Sensitive Information – Internal Use Only~~ 02-Apr-2019 **REGION 3 ENFORCEMENT STATUS REPORT**	
RIII PANEL	None
RIII RA/DRA PRE-BRIEFS	None
CONFERENCES	None
ALTERNATIVE DISPUTE RESOLUTION MEETINGS	None
OTHER NOTABLE MEETINGS	None

A red check mark (✔) in this report indicates a pending enforcement action is approaching or has exceeded a goal. Below are the timeliness goals associated with various types of enforcement actions.

Type of case	Timeliness goal	Clock starts
Based on OI report	Average case processing time shall be ≤180 days. No case shall exceed 330 days. OI report Issuance to Enforcement Panel: 45 days Enforcement Panel to Choice Letter Issuance: 30 days Choice Letter to PEC/ADR: 45 days (written 30 days) PEC/ADR/Written Response to Final Action: 30 days	Date of OI report
Standard Cases (Non-OI), (Non-SDP)	Average processing time shall be ≤120 days. No case shall exceed 160 days.	Exit Date
SDP Cases (Non-OI)	All enforcement actions issued ≤ 90 days. [NRR] Average processing time shall be ≤120 days. [OE] No case shall exceed 160 days.[OE]	Report Date (NRR) Exit Date (OE)
Disputed violations	Letter acknowledging receipt due to licensee within 30 days, if more than 30 days are necessary to review and respond. Final response to licensee shall be issued in ≤90 days, unless OE agrees to an extension.	Receipt of licensee's letter

REGION III ENFORCEMENT SUMMARY						April 2, 2019
DIVISION/ ADMIN	EA #	LICENSEE	ISSUE	DAYS		NEXT ACTION
DNMS	EA-18-157	Gerdau - Monroe Mill				
DNMS	EA-18-172	USS Silversides Submarine Museum				
DNMS	EA-19-007	Zevacor Molecular				
DNMS	EA-19-011	Air Zoo				
DNMS	EA-19-015	IRISNDT				
DNMS	EA-19-019	Wayne State University				
DNMS	EA-19-020	Jan X-Ray Services				
DNMS	EA-19-026	Army				(b)(5), (b)(7)(A)
[redacted]						

REGION III ENFORCEMENT SUMMARY						April 2, 2019
DIVISION/ ADMIN	EA #	LICENSEE	ISSUE	DAYS		NEXT ACTION
DRP	EA-18-104	Clinton	Failure to follow several procedures resulted in 2 EDGs unavailable at the same time.	221	90 (NRR) 11/21/18	Completed 4/1/19
				189	120 (OE) 1/22/19	
DRS	EA-15-206	Monticello	(b)(5), (b)(7)(A)			

1. MATERIALS CASES

DIVISION OF NUCLEAR MATERIALS SAFETY				
OE GOAL	LICENSEE NRC CONTACTS	EA #, IA #, OI #	ISSUE	RECENT & PENDING ACTIONS
DNMS 120 days: **05/04/19**	Gerdau - Monroe Mill DNMS: Craffey EICS: Lambert OE: Sreenivas OGC: Admin:	EA-18-157		(b)(5), (b)(7)(A)
	USS Silversides Submarine Museum DNMS: Lin EICS: Lambert OE: Holiday OGC: Admin:	EA-18-172		
DNMS 90 days: **04/10/19**	Zevacor Molecular DNMS: Draper EICS: Lambert OE: Marenchin OGC: Admin: Ind. Reviewers: Steve Bell/ Geoff Edwards	EA-19-007		
DNMS	Air Zoo DNMS: Learn/Lin EICS: Lambert OE: Marenchin OGC: Admin:	EA-19-011		
DNMS OI 180 days: **06/18/19**	IRISNDT DNMS: Null/Piskura EICS: Lambert OE: Furst OGC: Admin:	EA-19-015		

DIVISION OF NUCLEAR MATERIALS SAFETY

OE GOAL	LICENSEE NRC CONTACTS	EA #, IA #, OI #	ISSUE	RECENT & PENDING ACTIONS
DNMS	Wayne State Univ. DNMS: Craffey EICS: Lambert OE: Marenchin OGC: Admin: Clay	EA-19-019		(b)(5), (b)(7)(A)
DNMS 120 days: **xx/xx/18**	Jan X-Ray Serv. DNMS: Craffey EICS: Lambert OE: Marenchin OGC: Admin: Clay	EA-19-020		
DNMS	Army DNMS: Nieves EICS: Lambert OE: Marenchin OGC: Admin: Clay	EA-19-026		

2. REACTOR CASES (DRP)

DIVISION OF REACTOR PROJECTS				
OE & NRR GOALS	LICENSEE NRC CONTACTS	EA #, OI #, IA#	ISSUE	RECENT & PENDING ACTIONS
OE 120 days 01/22/19 NRR 90 days 11/21/18 NRR clock is limiting	Clinton DRP: Phillips DRS: SRA: Kozak EICS: Lambert OE: Marshfield Admin: Clay	EA-18-104	Failure to follow several procedures resulted in 2 EDGs unavailable at the same time.	05/17/18 Event date; PD determination date 07/19/18 IFRB held 07/26/18 Planning SERP held (sched. for 7/26) xx/xx/418 Complete detailed risk evaluation 09/20/18 SERP held (tent. sched. for 9/20) 09/24/18 Exit with licensee 10/10/18 draft preliminary white letter to OE and NRR 10/15/18 OE (10/12) and NRR (10/15) review/concurrence received 09/26/18 Strategy form signed 10/15/18 Inspection report/choice letter issued 10/19/18 Licensee written response re reg conference 11/13/18 Meeting notice issued 11/20/18 Pre-conference strategy session 11/30/18 Reg Conference held 12/14/18 Licensee provides additional written response (due 12/15) 02/14/19 Hold post-conference caucus (Final SERP) (2/14) 02/28/19 Hold follow-up post conference caucus (2/28) 03/14/19 DRP provide technical input for final action 03/19/19 EICS provide final action to Admin 03/20/19 Admin place final action into concurrence 03/21/19 Draft action forwarded to OE and NRR 03/26/19 OE and NRR review/concurrence received 03/26/19 EN issued 04/01/19 Final action issued **ACTION:** Completed 4/1/19

3. REACTOR CASES (DRS)

DIVISION OF REACTOR SAFETY				
OE & NRR GOALS	LICENSEE NRC CONTACTS	EA #, OI #	ISSUE	RECENT & PENDING ACTIONS
OE 90 days	Monticello DRS: D. Hills EICS: Lambert OE: Marshfield	EA-15-206		(b)(5), (b)(7)(A)

4. ORDERS & CONFIRMATORY ACTION LETTERS

ORDERS					
Division	**EA#, IA#**	**Licensee**	**Order Description**	**Open or Completed**	**Notes**
DNMS	EA-14-193	Monticello			
DNMS	EA-16-282	Tilden Mining			
DRP	EA-15-039	Palisades			
DRS	EA-16-022	Davis-Besse			

(b)(5), (b)(7)(A)

ORDERS

Division	EA#, IA#	Licensee	Order Description	Open or Completed	Notes
Individual	IA-18-043	Mistras			
Individual	IA-16-049 Order dated June 1, 2017	JANX			
Individual	IA-16-059 Order dated February 2, 2017	American Engineering Testing			
Individual	IA-14-039 Order dated August 4, 2015	Cardiology II, P.C.			
Individual	IA-09-035	Philadelphia Veterans Affairs Medical Center			
Individual	IA-10-010	Philadelphia Veterans Affairs Medical Center			
Individual	IA-13-012	University Nuclear and Diagnostics, LLC			
Individual	IA-13-024	Dresden			

(b)(5), (b)(7)(A)

ORDERS					
Division	**EA#, IA#**	**Licensee**	**Order Description**	**Open or Completed**	**Notes**
Individual	IA-13-025	Dresden	(b)(5), (b)(7)(A)		

Note to requester: A draft version of EA-18-104 is withheld in its entirety under FOIA exemption B5 (deliberative process priviledge).

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Enclosure

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SHUTDOWN OPERATIONS
SIGNIFICANCE DETERMINATION PROCESS
PHASE 1 INITIAL SCREENING AND CHARACTERIZATION OF FINDINGS

1.0 APPLICABILITY

This attachment and its exhibits are designed to provide U.S. Nuclear Regulatory Commission (NRC) inspectors and management with a framework for use in the initial screening and characterization of potentially risk-significant Shutdown (SD) issues within the Initiating Events, Mitigation Systems, and Barrier Integrity cornerstones for the Significance Determination Process (SDP). In addition, this process identifies findings of very low risk significance that do not warrant further NRC engagement. This appendix is intended to be used when the plant is shutdown with at least one fuel bundle in the reactor and temperature and pressure are within the normal Residual Heat Removal (RHR)/ Decay Heat Removal (DHR) conditions, otherwise return to IMC 0609, Attachment 4, "Initial Characterization of Findings."

2.0 ENTRY CONDITIONS

Before entering an issue into the SDP, the inspector will screen the issue to determine its documentation threshold as described in IMC 0612, Appendix B, "Issue Screening. If an inspector screens a finding in accordance with Appendix B and is directed by Appendix B to determine its risk significance, and if that finding involves shutdown operations with fuel in the reactor, then the inspector will initially screen that finding using the SD Phase 1 screening questions found in Exhibits 2-5.

3.0 PHASE 1 SDP OVERVIEW

Appendix G of the SDP is a tool which uses a quantitative risk method to characterize the risk of events or conditions during SD. All issues, including those at SD, that screen more than minor in Appendix B of IMC 0612 are then characterized using IMC 0609, Attachment 4. If the finding impacts the Initiating Events, Mitigation Systems, or Barrier Integrity Cornerstone, then Table 3 in IMC 0609, Attachment 4 will refer the inspector to the appropriate SDP Appendix. In the case of a SD finding, it will refer them to this appendix. The inspector would utilize the information from their initial characterization of the finding in IMC 0609, Attachment 4, Tables 1 & 2, but would transfer to this Appendix in Step A of Table 3 when directed by IMC 0609, Attachment 4. The purpose of the screening questions in Exhibits 2-5 is to determine if the issue can be characterized as Green before entering into a more detailed analysis with IMC 0609, Appendix G Phase 2 or 3.

Phase 1 is intended to be accomplished by the inspection staff, with the assistance of a Senior Reactor Analyst (SRA), if needed. Inspectors should collect information needed for determining the significance of the finding, such as the structure, system, or component affected, the nature of the degradation, and the duration of the degraded condition. Inspectors should obtain licensee risk perspectives as early in the SDP process as a licensee is prepared to offer them, and use the SDP framework to the extent possible to evaluate the adequacy of the licensee's input and assumptions.

END

Exhibit 1 – User Guidance for Appendix G Phase 1: Initial Screening and Characterization of Findings

Exhibit 2 – Initiating Events Screening Questions

Exhibit 3 – Mitigating Systems Screening Questions

Exhibit 4 – Barrier Integrity Screening Questions

Exhibit 5 – External Events Screening Questions

Exhibit 1 - User Guidance for Appendix G Phase 1: Initial Screening and Characterization of Findings

Step 1: Perform an initial screening of the inspection finding.

CAUTION: Most shutdown finding risk results are driven by the operator failure probabilities. When evaluating shutdown findings it is important to be aware of any conditions or events that may impact operator response.

1.1 It is important to note that current fleet Pressurized Water Reactor (PWR) designs do not have automatic safety actuation systems during shutdown. Also, in the current Boiling Water Reactor (BWR) designs there is no requirement to have the automatic low level injection initiation functional in cold shutdown and refueling. Therefore, the risk significance of many findings will rely on operator's ability to diagnose the problem and perform appropriate actions. Successful operator actions are dependent on plant procedures, available time, complexity of the mitigation response, training, ability to diagnose the problem, etc. Therefore, when evaluating the initial screening of a shutdown finding it is important to be aware of any conditions or events that may impact the operators' ability to diagnose and respond to a shutdown initiator. If you have any questions or are uncertain about an issue you are evaluating contact your Regional SRA.

1.2 Table G1 provides an overview of key safety functions and systems important to safety during shutdown, the inspector should use this table while completing the appropriate Exhibit 2-5. This table attempts to collect all potential influences on both the human actions and equipment that can affect the risk at shutdown. Inspectors should use the information in Table G1 to determine which, if any, categories of Exhibits 2-5 that are influenced by specific findings.

1.3 If the finding affects the safety of a reactor at shutdown, **THEN IDENTIFY** the affected cornerstone(s):

- ☐ Initiating Event
- ☐ Mitigation Systems
- ☐ Reactor Coolant System (RCS) Barrier
- ☐ Fuel Barrier
- ☐ Containment Barriers

NOTE: When assessing the significance of a finding affecting multiple cornerstones, the finding should be assigned to the cornerstone that best reflects the dominant risk of the finding.

CONTINUE to the appropriate Exhibit 2-5 to answer the screening questions.

1.4 Use the decision logic in the exhibits when answering the screening questions to determine if the issue can be characterized as Green. Note that the examples

provided in the exhibits are not all inclusive. If you have any questions or are uncertain about an issue you are evaluating contact your Regional SRA.

Step 2: If the finding screens as Green, then document in accordance with IMC 0612.

Step 3: If the finding screens as other than Green, perform an Appendix G Phase 2 or Phase 3 analysis as directed by the screening questions in Exhibits 2-5

Table G1 Generic SD Key Safety Functions and System Dependencies[1]			
Safety Function	**Major Systems**	**Supporting Systems**	**Initiating Event Scenarios**
Decay Heat Removal	• Residual Heat Removal • Decay Heat Removal • Shutdown Cooling • Steam Generators (PWR) • Feed and Bleed (Low Pressure Injection, High Pressure Injection, Charging System (PWR) • Control Rod Drive System (BWR) • Core Spray(BWR)	• AC Power • DC Power • RHR/DHR Heat Exchanger • Component Cooling Water (PWR) • Power Operated Relief Valves (PWR) • Instrumentation (i.e., RCS Level, RHR/DHR Heat Exchanger inlet/outlet Temperature and RHR/DHR Flow Indication, Core Exit Thermocouples (PWRs with reactor head installed only) • Residual Heat Removal Service Water (BWR) • Safety Relief Valves (BWR) • Training • Procedures • Time to Boil and Time to Core Uncovery	• Loss of RHR (LORHR) • Loss of SDC (LOSDC) • Loss of Off-site Power (LOOP) • Loss of Inventory (LOI) • Overdrain (OD) (PWR) • Loss of Level Control[2] (LOLC) (PWR) • Loss of Component Cooling Water (CCW) (PWR) • Loss of Residual Heat Removal Service Water (RHRSW) (BWR)

1 This table is not intended to be all-inclusive. It is intended to give the inspector an overview of important systems and key safety functions to consider when characterizing the SD finding.

Table G1 Generic SD Key Safety Functions and System Dependencies[1]			
Safety Function	**Major Systems**	**Supporting Systems**	**Initiating Event Scenarios**
Inventory Control	• Low Pressure Injection • High Pressure Injection • Charging System (PWR) • Control Rod Drive System (BWR) • Core Spray (BWR)	• Drain Down Isolation Valve(s) • AC Power • DC Power • RHR/DHR Heat Exchanger • RHR/DHR Relief Valves • Power Operated Relief Valves (PWR) • Instrumentation (i.e., RCS Level, RHR/DHR Heat Exchanger inlet/outlet Temperature and RHR/DHR Flow Indication, Core Exit Thermocouples (PWRs with reactor head installed only) • Safety Relief Valves (BWR) • Training • Procedures • Time to Boil and Time to Core Uncovery	• Loss of Inventory (LOI • Overdrain (OD) (PWR) • Loss of Level Control[2] (LOLC) (PWR)
Electric Power Availability	• Emergency Diesel Generators • Offsite Power Feeds • Offsite Transformers • Offsite Inverters	• AC and DC Busses • Batteries and Battery Charges • Motor Generators • Inverters • Training • Procedures • Time to Boil and Time to Core Uncovery	• All Initiators

2 Loss of level control requires a Phase 2 or Phase 3 if
Loss of Level for PWRs: inadvertent loss of 2 feet of RCS inventory when not in mid-loop OR
inadvertent entry into reduced inventory OR
mid-loop conditions OR
inadvertent loss of 2 inches of RCS inventory when in mid-loop conditions.
Loss of Level for BWRs: inadvertent loss of 2 feet of RCS inventory OR
inadvertent RCS pressurization.

Table G1 Generic SD Key Safety Functions and System Dependencies[1]			
Safety Function	**Major Systems**	**Supporting Systems**	**Initiating Event Scenarios**
Reactivity Control	• RPS • Control rod and associated drive mechanisms • Chemical and Volume Control System (PWR) • Standby Liquid Control (BWR)	• AC Power • DC Power • Nuclear Instrumentation • Training • Procedures • Time to Boil and Time to Core Uncovery	• Reactivity (inadvertent criticality)
Containment	• Hydrogen Control • Containment Closure Capability • Penetrations	• AC Power • DC Power • Motive Power to close Hatches (assuming loss of AC power) • Temporary closures/penetrations • Training • Procedures • Time to Boil and Time to Core Uncovery	• All Initiators

Exhibit 2 - Initiating Events Screening Questions

A. SD Initiators

1. Does the finding increase the likelihood of a SD initiating event?

   □ **If YES ➛ Stop. Go to Appendix G Phase 2.**

   □ If NO, continue.

B. Loss of Coolant Accident - Loss of Inventory (LOI) Initiators

2. Did a LOI event result in a leakage such that if the leakage were undetected and/or unmitigated in 24 hours or less it would cause the currently operating decay heat removal method to fail (e.g., level would drop to below the hotleg suction of the operating decay heat removal pump (PWR), or to the shutdown cooling isolation Level 3 setpoint (BWR))?

   □ **If YES ➛ Stop. Go to Appendix G Phase 2.**

   □ If NO, continue.

3. Is the LOI event self-limiting such that leakage will stop before impacting the operating method of decay heat removal?

   □ If YES, continue.

   □ **If NO ➛ Stop. Go to Appendix G Phase 2**.

C. Transient Initiators

4. LOOP - Did the initiator occur when refuel canal/cavity was flooded?

   □ If YES, continue.

   □ **If NO ➛ Stop. Go to Appendix G Phase 2.**

5. LOOP - Did the initiator occur when the time to boil off RCS inventory to the top of active fuel (TAF) was shorter than the time to recover offsite power?

   □ **If YES ➛ Stop. Go to Appendix G Phase 2.**

   □ If NO, continue.

6. LORHR - Did the initiator occur when refuel canal/cavity was flooded?

- □ If YES, continue.
- □ **If NO ➛ Stop. Go to Appendix G Phase 2.**

7. Loss of Level Control (LOLC) or Over Drain (OD) - For PWRs, did the initiator occur when reactor level was in reduced inventory?

- □ **If YES ➛ Stop. Go to Appendix G Phase 2.**
- □ If NO, continue.

D. <u>External Event Initiators</u>

8. Does the finding increase the likelihood of a fire or internal/external flood that could cause an SD initiating event?

- □ **If YES ➛ Stop. Go to Phase 3.**
- □ If NO, screen as Green.

Exhibit 3 – Mitigating Systems Screening Questions

A. Mitigating Structure System Component (SSC) and Functionality

1. If the finding is a deficiency affecting the design or qualification of a mitigating SSC, does the SSC maintain its operability or functionality?

   - ☐ **If YES, screen as Green.**
   - ☐ If NO, continue.

2. Does the finding represent a loss of system safety function?

   - ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
   - ☐ If NO, continue.

3. Does the finding represent an actual loss of safety function of at least a single Train for greater than its Tech Spec Allowed Outage Time, OR two separate safety systems out-of-service for greater than its Tech Spec Allowed Outage Time?

   - ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
   - ☐ If NO, continue.

4.a) If the cavity is flooded, does the finding represent an actual loss of safety function of one or more non-Tech Spec Trains of equipment during SD designated as risk-significant (e.g. 10CFR50.65), for greater than 24 hrs?

   - ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
   - ☐ If NO, continue.

4.b) If the cavity is not flooded, does the finding represent an actual loss of safety function of one or more non-Tech Spec Trains of equipment during SD designated as risk-significant (e.g. 10CFR50.65), for greater than 4 hrs?

   - ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
   - ☐ If NO, continue.

5.a) For PWRs, does the finding degrade RCS level indication and/or core exit thermal couples (CETs) when the cavity is not flooded?

   - ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
   - ☐ If NO, continue.

5.b) For BWRs, does the finding degrade a functional auto-isolation, regardless of whether it is required to be operable or not, of RHR on low reactor vessel level?

- ☐ **If YES ➛ Stop. Go to Appendix G Phase 2.**
- ☐ If NO, continue.

B. <u>External Event Mitigation Systems (Seismic/Fire/Flood/Severe Weather Protection Degraded)</u>

6. Does the finding screen as potentially risk significant due to a seismic, flooding, or severe weather initiating event, using the criteria in Exhibit 5?

- ☐ **If YES ➛ Stop. Go to Phase 3.**
- ☐ If NO, continue.

C. <u>Fire Brigade</u>

7. Does the finding involve Fire Brigade training and qualification requirements, or brigade staffing?

- ☐ **If YES ➛ check if one or more of the following apply:**
  - The fire brigade demonstrated the ability to meet the required times for fire extinguishment for the fire drill scenarios, and the finding did not significantly affect the ability of the fire brigades to respond to a fire.
  - The overall time duration (exposure time) that the Fire Brigade was understaffed was short (< 2 hours).

  - ☐ **If none of the above apply ➛ Stop. Go to IMC 0609, Appendix M.**
  - ☐ **If at least one of the above is applicable, continue.**
- ☐ If NO, continue.

8. Does the finding involve the response time of the Fire Brigade to a fire?

   □ **If YES ➛ check if one or more of the following apply:**
   - The fire brigade's response time was mitigated by other defense-in-depth elements, such as area combustible loading limits were not exceeded, installed fire detection systems were functional, and alternate means of safe shutdown were not impacted.
   - The finding involved risk-significant fire areas that had automatic suppression systems.
   - The licensee had adequate fire protection compensatory actions in place.

     □ **If none of the above apply ➛ Stop. Go to IMC 0609, Appendix M.**

     □ **If at least one of the above is applicable, continue.**

   □ If NO, continue.

9. Does the finding involve fire extinguishers, fire hoses, or fire hose stations?

   □ **If YES ➛ check if one or more of the following apply:**
   - There was no degraded fire barrier and the fire scenario did not require the use of water to extinguish the fire.
   - The missing fire extinguisher or fire hose was missing for a short time and other extinguishers or hose stations were in the vicinity.

     □ **If none of the above apply ➛ Stop. Go to IMC 0609, Appendix M.**

     □ **If at least one of the above is applicable, screen as Green.**

   □ **If NO, screen as Green.**

Exhibit 4 – Barrier Integrity Screening Questions

A. <u>RCS or Fuel Barrier</u>

Note: If the finding involves fuel bundle misplacement or misorientation in the reactor core, **screen as Green.**

1. Low Temperature Over Pressurization (LTOP) – For PWRs, does the finding involve an inadvertent Safety Injection Actuation, the unavailability of a PORV or LTOP relief valve or their associated setpoints during LTOP operations or when it is required?

   □ **If YES ➛ Stop. Go to Phase 3.**

   □ If NO, continue.

2. Freeze Seal – Does the finding increase the potential for failure of the freeze seal or if unmitigated have the potential to cause a disruption in RHR/DHR or a LOI event?

   □ **If YES ➛ Stop. Go to Phase 3.**

   □ If NO, continue.

3. Steam Generator (SG) Nozzles Dams – Does the finding involve improper SG nozzle dam installation (e.g. hot leg manway must be opened first, hotleg SG nozzle dam installed last), inadequate SG nozzle dam RCS vent path, deficiencies of the SG nozzle dams (Ref GL 88-17 and IN 88-36) or SG nozzle dam functionality?

   □ **If YES ➛ Stop. Go to Phase 3.**

   □ If NO, continue.

4.a) Criticality – For PWRs, does the finding involve the potential for, or an actual, RCS boron dilution event?

   □ **If YES ➛ Stop. Go to IMC 0609, Appendix M.**

   □ If NO, continue.

4.b) Criticality – For BWRs, does the finding involve 2 or more adjacent control rods with the potential to, or actually, add positive reactivity?

   □ **If YES ➛ Stop. Go to IMC 0609, Appendix M.**

   □ If NO, continue.

5. Drain Down Path or Leakage Path - Does the finding degrade the ability to isolate a drain down or leakage path?

   □ **If YES ➛ Stop. Go to Phase 3.**

   □ If NO, continue.

B. <u>Containment Barrier</u>

6. Does the finding degrade the ability to close or isolate the containment (this includes but is not limited to equipment and personnel hatches and permanent and temporary penetrations)?

   □ **If YES ➛ Stop. Go to IMC 0609, Appendix H.**

   □ If NO, continue.

7. Does the finding degrade the physical integrity of reactor containment (valves, penetrations, containment isolation components)?

   □ **If YES ➛ Stop. Go to IMC 0609, Appendix H.**

   □ If NO, continue.

8. Does the finding involve an actual reduction in function of hydrogen control for BWR Mark III and PWR ice condenser containments?

   □ **If YES ➛ Stop. Go to IMC 0609, Appendix H.**

   □ If NO, screen as Green.

Exhibit 5 – External Events Screening Questions

1. If the equipment or safety function is assumed to be completely failed or unavailable, are ANY of the following three statements TRUE? The loss of this equipment or function by itself, during the external initiating event it was intended to mitigate:

    - would cause any of the Initiating Events used by Table G1 for the plant in question;
    - would degrade **two or more** trains of a multi-train safety system or function, or would degrade the only available train, which would defeat the entire safety function;
    - would degrade one or more trains of a system that supports a safety system or function.

    ☐ **If YES** ➛ the finding is potentially risk significant due to external initiating event core damage sequences return to screening questions in Exhibits 2-5.

    ☐ If NO, continue.

2. Does the finding involve the total loss of any safety function, identified by the licensee through a Probalistic Risk Assessment, Individual Plant Examination External Events, or similar analysis, that contributes to external event initiated core damage accident sequences (i.e., initiated by a seismic, flooding, or severe weather event)?

    ☐ **If YES** ➛ the finding is potentially risk significant due to external initiating event core damage sequences return to screening questions in Exhibits 2-5.

    ☐ If NO, screen as Green.

Attachment 1
Revision History Page

Commitment Tracking Number	Accession Number Issue Date Change Notice	Description of Change	Description Training Required And Completion Date	Comment Feedback Resolution Accession Number
N/A	05/25/04 CN 04-015	Initial issuance	N/A	N/A
	ML13050A934 05/09/14 CN 14-011	IMC 0609 App G, Att. 1 is revised to enhance the usability of this appendix, based on feedback received from the SRA. The formatting was updated to be consistent with IMC 0609 Appendix A. The checklists from the previous revision, for PWRs and BWRs, were combined into one list in the various Exhibits in the attachment using screening questions and decision logic. The content was updated and reworked to be more user-friendly for inspectors to screen findings to determine if they are Green or a more detailed analysis is needed. Incorporated feedback from ROPFF 0609G1-1911 and 0609G-1323. This is a complete reissue no red line.	N/A	ML13162A640 0609G-1323 ML14120A177 0609G1-1911 ML1412A166

Human Failure Event (HFE) ID: SD-EPS-XHE-XM-NR01H

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EXHIBIT 1 – IFRB FINDING FORM

IFRB Cover Sheet	
Facility Name/Location: Clinton	Name of Utility or Licensee: Exelon
Docket Number(s): 50-461	EA Number: EA-18-104

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