Turkey Point Nuclear Plant Unit 3 and Unit 4
License Amendment Request 276, Revise Fire Protection Program in Support of Reactor Coolant Pump Seal Replacement Project

ATTACHMENT 2

APPLICATION SPECIFIC FIRE PRA MODEL TO SUPPORT LAR AMENDMENT FOR REPLACEMENT OF RCP SEALS AT TURKEY POINT UNITS 3 AND 4

(NON-PROPRIETARY VERSION)

(70 pages follow)

REPORT

APPLICATION SPECIFIC FIRE PRA MODEL TO SUPPORT LAR AMENDMENT FOR REPLACEMENT OF RCP SEALS

Turkey Point Units 3 and 4

REVISION 1

FRAMATOME INC.

This document has been redacted and no longer contains Framatome Proprietary Information.

This redacted version is Non-Proprietary.

PREPARED FOR

Stuart Brown, Framatome
10101 David Taylor Drive
Suite 200
Charlotte, NC 28262

Project #: 1UAF6S001
Report #: 06S001-RPT-01
Date: August 19, 2022

PREPARED BY

Usama Farradj
1220 Concord Avenue, Suite 400
Concord, CA 94520 USA

UFarradj@jensenhughes.com
+1 925-938-3550

Preparer:	Jazmin NeVille
Reviewer	Nate Pratt
Review Method	☒ Design Review ☐ Alternate Calculation
Approver	Usama Farradj

Revision Record Summary

Revision	*Revision Summary*
0	Initial use (issued 8/12/22)
1	Replace Assumptions Requiring Validation with issued references, editorial changes, elimination of Appendices A through I (due to large volume of data, available via project files)

Table of Contents

1.0 Introduction

This application specific analysis is being performed to evaluate the risk impact of replacement of the current Turkey Point (PTN) Reactor Coolant Pump (RCP) seals with the Framatome Passive Shut Down Seal (PSDS).

2.0 Purpose

This risk analysis focuses on the Fire PRA (Probabilistic Risk Assessment) risk impact of the use of the Framatome PSDS configuration and evaluates the associated risk increase. This supports a License Amendment Request for NRC approval of a risk increase that exceeds the NFPA 805 license condition limit for self-approval of fire protection program changes.

3.0 Technical Evaluation

3.1 FIRE PROBABILISTIC ASSESSMENT

The Fire PRA is updated to incorporate a model of the Framatome PSDS seal in conjunction with Fire PRA model refinements to incorporate recent EPRI Fire PRA realism NUREGs.

3.2 SUMMARY OF PRA APPROACH

Section 3.6.2 provides a detailed discussion of the PRA model logic changes incorporated to address the PSDS design configuration. Additional refinements to the Fire PRA have been performed to provide a more realistic assessment of the plant risk and the delta risk associated with this modification. These refinements include the interruptible fire modeling defined in NUREG-2230, “Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinet Fires in Nuclear Power Plants” [1], updated transient heat release rates defined in NUREG-2233, “Methodology for Modeling Transient Fires in Nuclear Power Plant Fire Probabilistic Risk Assessment” [2] as well as guidance provided in NUREG-2178, Volume 2, “Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire” [3] regarding a more realistic assessment of the impact of fires near walls and corners.

The status of Full Power Internal Events (FPIE)/Internal Flooding (IF)/Fire PRA peer reviews and Findings and Observations (F&O) closure activities is summarized in Appendix B.

3.3 SUMMARY OF FIRE PRA METHODS INCORPORATED IN THE RCP SEAL REPLACEMENT QUANTIFICATION

The changes performed to model the Framatome PSDS configuration are considered updates to the PRA model given that they use the existing logic and refine it to allow assessment of PSDS model specific failure modes.

The use of NUREG-2230, NUREG-2233 and NUREG-2178, Volume 2 as discussed in the previous section are also considered to be PRA model updates and not new methods or upgrades given their application of methodologies similar to those applied in the original model. Primary changes are associated with fire suppression event tree structure, heat release rate applicability and wall and corner impact criteria relaxation.

Based on the above, the replacement of the RCP seal is considered a PRA maintenance level update not an upgrade.

3.4 CUMULATIVE RISK OF CHANGES

Cumulative risk incurred subsequent to the final NFPA 805 model is primarily associated with reductions in risk associated with model refinements. No plant modifications exceeding a no more than minimal/negligible risk increase were implemented since the final NFPA 805 post transition model.

3.5 SUMMARY OF DATA USED TO SUPPORT THE FIRE PRA

The primary input data, other than the fault tree logic changes, discussed in Section 3.6.2, are the RCP seal failure probabilities and the system time window for RCP trip operator actions.

The total Framatome PSDS RCP seal failure probability is [ ]

This failure probability is the sum of the failure probabilities associated with the following seal failure modes:

+ Failure to Actuate (FTA) – [ ] [6]
+ Failure to Remain Sealed (FTRS) – [ ] [7]
+ Spurious Actuation (SA) – [ ] [8]

3.6 ANALYSIS – FIRE PRA MODEL CHANGES TO MODEL OF RECORD

3.6.1 Plant Partitioning/Ignition Frequency Report Changes

No plant partitioning or ignition frequency changes were incorporated in the model update for the replacement of the RCP seals.

3.6.2 Component/Cable Report Changes

This Application Specific Model (ASM) takes as a baseline for comparison, the last issued Fire PRA fault tree - Revision 14F - documented under revision 7 of the component/cable report [4].

3.6.2.1 Fault Tree Logic Changes

The ASM/modified fault tree added 20 basic events and removed 45. The scope and concerns addressed by the changes is limited. The changes to the ASM/modified fault tree center around a few distinct areas detailed below.

RCP Seal Individual Modeling

> The ASM fault tree now models each of the six RCP seals (three per unit) present at the site explicitly and individually. As part of this refinement, RCPs and their associated steam generators (and further, individualized flowpaths) are also now modeled individually and explicitly in the ASM/modified fault tree.
>
> The failure modes of the PSDS also differ somewhat from those considered for the current RCP seals. The ASM/modified fault tree was changed to accurately model the new behavior and failure modes of the new seals.

The primary changes to the fault tree included:

1. Incorporation of a failure mode associated with asymmetric cooling of the RCS due to loss of auxiliary feedwater to an RCP loop
2. Incorporation of a failure mode associated with re-initiation of seal cooling after an initial loss of seal cooling which could result in thermal shock to the seals
3. Isolation of RCP No. 1 seal leakoff path resulting in a significant reduction in the system time window for RCP trip which would make an operator action not feasible and therefore would lead directly to seal failure if all seal cooling is unavailable [5]. The RCP No. 1 seal leakoff line isolation valves, CV-3/4-

303A/B/C, are modified to provide a double break circuit configuration during normal operation to reduce the impact on seal failure.

The figures below provide the key logic for the replacement RCP seals as depicted in the CAFTA fault tree model.

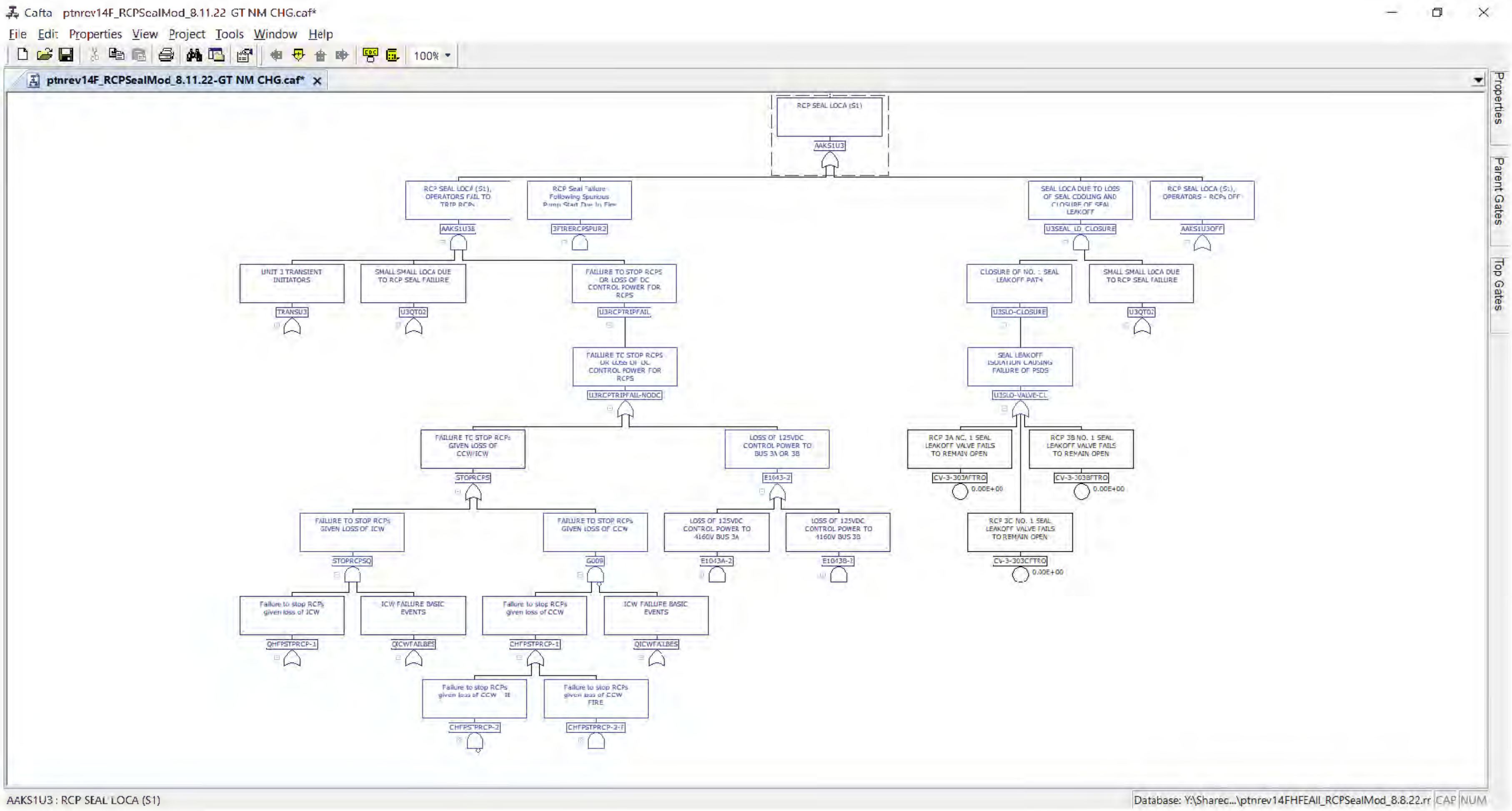


Figure 3-1 RCP Seal Replacement Fault Tree Modifications

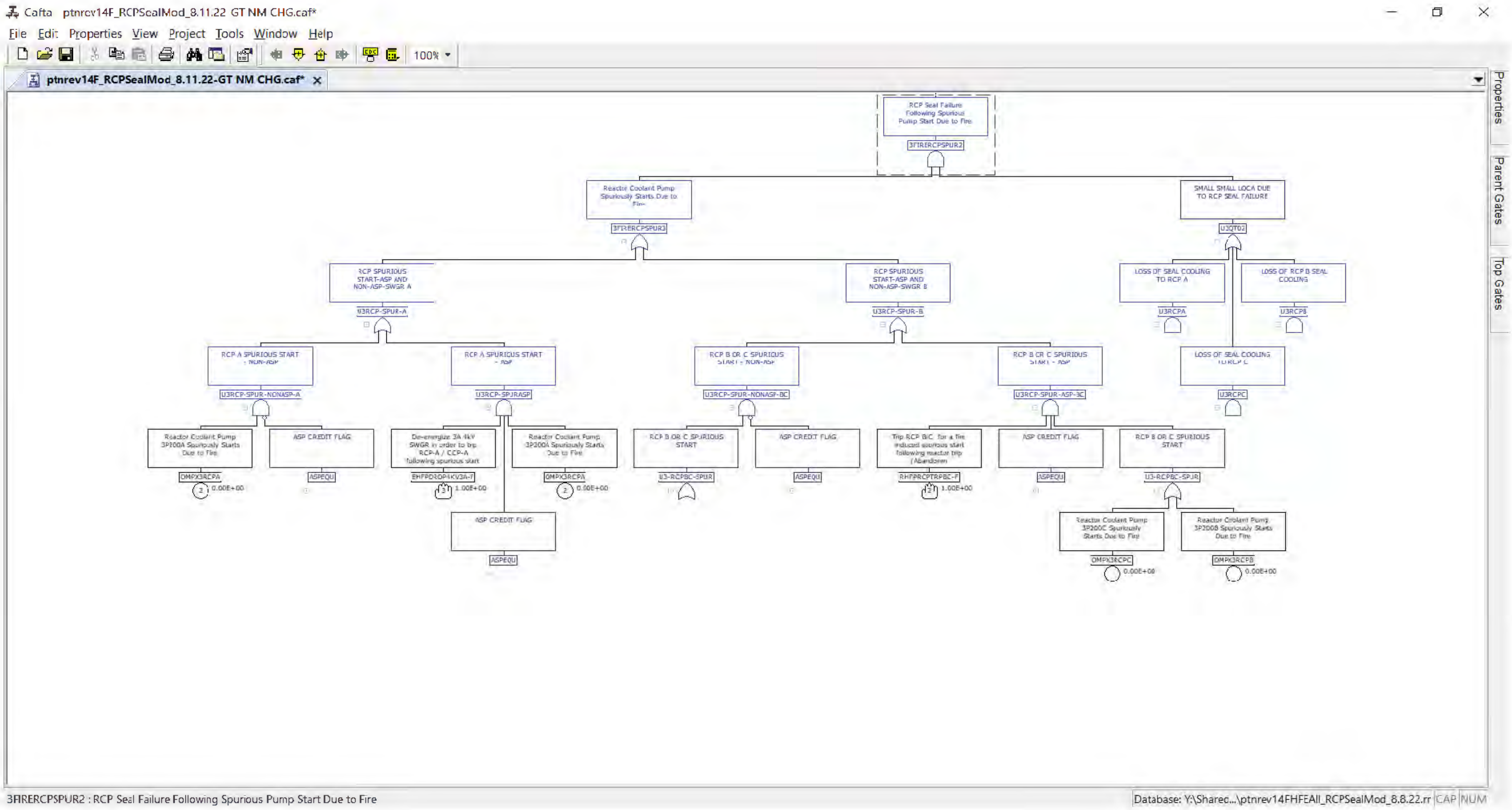


Figure 3-1 RCP Seal Replacement Fault Tree Modifications (continued)

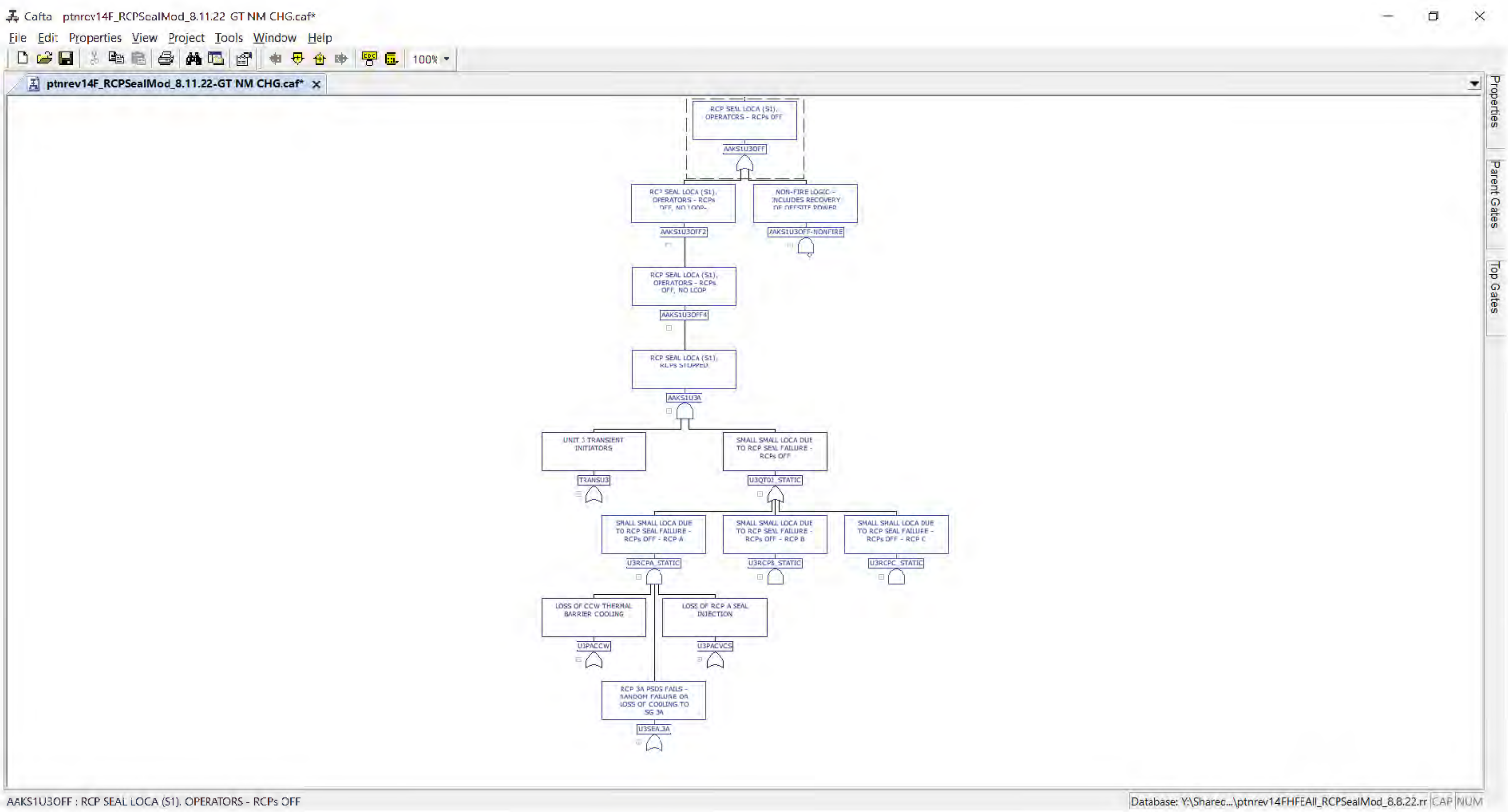


Figure 3-1 RCP Seal Replacement Fault Tree Modifications (continued)

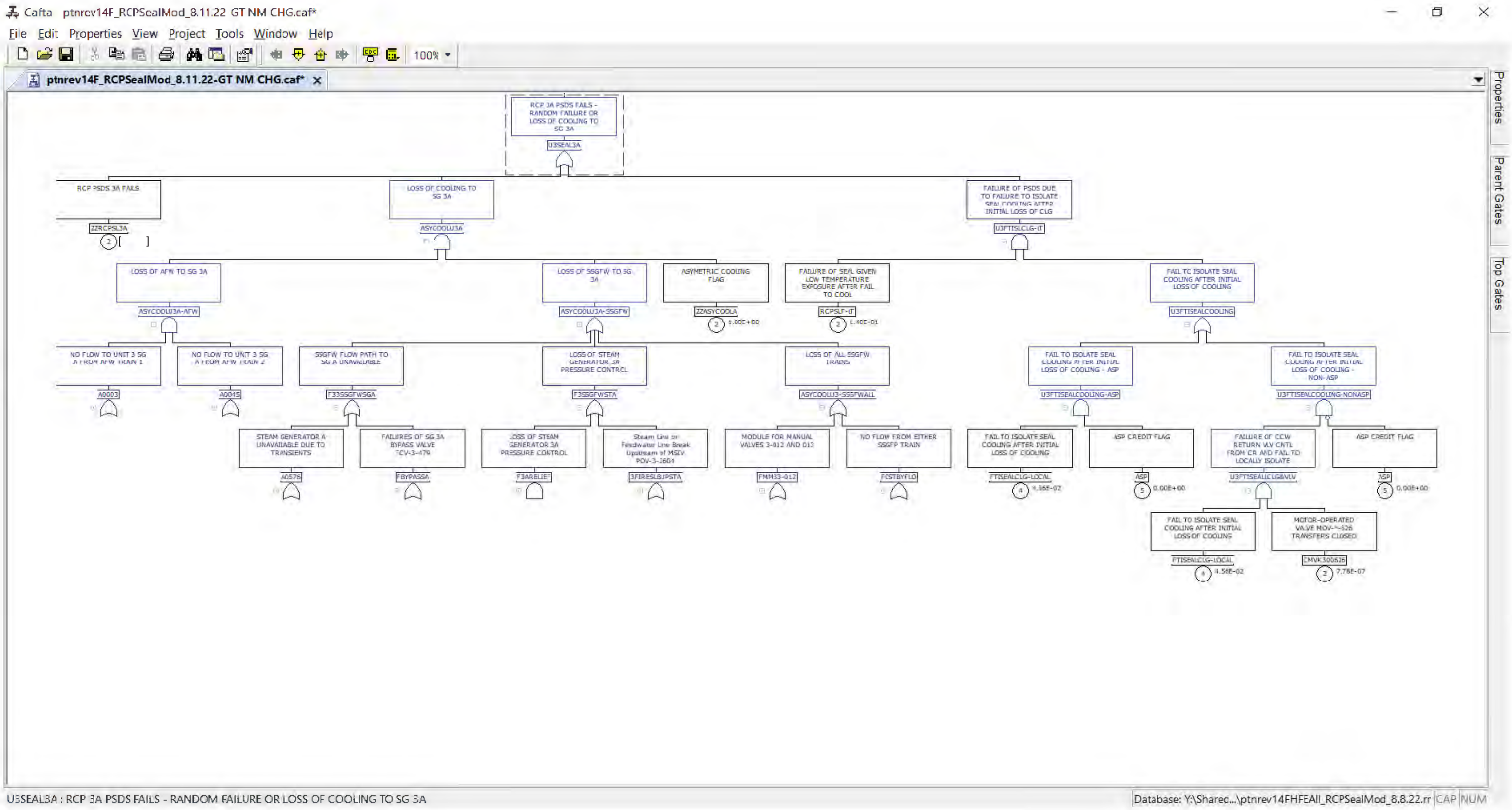


Figure 3-1 RCP Seal Replacement Fault Tree Modifications (continued)

3.6.2.2 Basic Event Mapping Changes

Changes to the component – basic event mapping were implemented to correspond to the seal model changes discussed above.

3.6.2.3 Component-Cable Mapping changes

No component-cable mapping changes of existing component-cable relationships was required. No new circuit analysis was performed to create or remove component-cable relationships. The cable routing for the No. 1 seal leak-off valves contained in prior versions of this model were restored to support implementation of the Framatome PSDS. Mapping of the associated BEs to the component/cable/raceway data was implemented to support the No. 1 Seal Leakoff logic specified above.

3.6.3 Multi-Compartment Analysis/Hot Gas Layer Scenario Report Changes

Changes to hot gas layer (HGL) and multi-compartment analysis (MCA) results are associated with incorporation of NUREG-2230 for interruptible fires, NUREG-2233 for transient fire heat release rates and combining the methods of NUREG-2230 and NUREG-2180 for the cable spreading room in-panel detection systems. See Sections 3.6.4 for a more detailed discussion of these changes.

The table below provides the summary of MCA non-suppression probabilities (NSP) which have been revised to reflect NUREG-2230 and NUREG-2233. MCA scenarios that damaged all cables in the zone but did not contribute to the HGL were given a conservative NSP of 1E-9, these scenarios are not included in the table below. This NSP was sufficiently low to remove these scenarios from significant risk contribution while documenting the associated fire zone MCA interaction.

Table 3-1 - MCA Scenarios

Scenario	*NSP*
020-MCA-2-PTB	2.86E-02
020-MCA-3-PTB	2.81E-02
021-MCA-1-PTB	3.45E-02
021-MCA-2-PTB	3.12E-02
021-MCA-3-PTB	3.12E-02
021-MCA-4-PTB	1.42E-02

Table 3-1 - MCA Scenarios

Scenario	NSP
021-MCA-5-PTB	1.66E-02
022-MCA-1-PTB	1.97E-02
022-MCA-2-PTB	1.78E-02
022-MCA-3-PTB	1.78E-02
022-MCA-4-PTB	7.94E-03
022-MCA-5-PTB	9.38E-03
026-MCA-2-PTB	8.98E-03
028-MCA-7-PTB	7.35E-03
061-MCA-1-PTB	8.37E-02
061-MCA-2-PTB	6.16E-04
061-MCA-3-PTB	3.47E-02
062-MCA-1-PTB	1.11E-01
062-MCA-2-PTB	3.54E-03
062-MCA-3-PTB	1.07E-01
062-MCA-4-PTB	1.08E-01
063-MCA-2-PTB	5.24E-02
063-MCA-3-PTB	5.17E-02

Table 3-1 - MCA Scenarios

Scenario	NSP
063-MCA-4-PTB	8.32E-03
065-MCA-1-PTB	6.77E-02
065-MCA-2-PTB	2.27E-01
067-MCA-1-PTB	1.13E-01
067-MCA-2-PTB	1.65E-01
067-MCA-3-PTB	1.59E-01
068-MCA-1-PTB	6.73E-02
068-MCA-2-PTB	1.16E-01
068-MCA-3-PTB	1.10E-01
070-MCA-1-PTB	8.55E-02
070-MCA-2-PTB	1.19E-01
070-MCA-3-PTB	1.18E-01
071-MCA-1-PTB	6.73E-02
071-MCA-2-PTB	1.13E-01
071-MCA-3-PTB	1.10E-01
101-MCA-1-PTB	2.04E-04
101-MCA-2-PTB	7.74E-03

Table 3-1 - MCA Scenarios

Scenario	NSP
101-MCA-3-PTB	2.10E-02
101-MCA-4-PTB	8.63E-03
101-MCA-5-PTB	6.22E-04
108A-MCA-1-PTB	7.95E-04
108A-MCA-2-PTB	1.67E-03
134-MCA-1-PTB	1.23E-02
134-MCA-2-PTB	9.74E-03
135-MCA-1-PTB	1.67E-02
135-MCA-3-PTB	1.67E-02
140-MCA-1-PTB	9.74E-03
140-MCA-2-PTB	1.63E-02

3.6.4 Scenario Report Changes

3.6.4.1 NUREG-2230 Incorporation into the Fire PRA via the Fire Modeling Workbook (FMW)

The scenario report is updated to reflect the changes made for updating the Non-Suppression Probability calculation from use of the SDC Tool to the Fire Modeling Workbook. Appendix D describes the changes made to implement the Fire Modeling Workbook and the NUREG-2230 interruptible fire scenario refinements.

3.6.4.2 NUREG-2230/2180 Methodology

The Incipient Detection credited in the cable spreading room is modified to include NUREG-2230 [1] along with NUREG-2180 [9]. This has been done for the cable spreading room which is provided with an in-panel incipient detection system for many of the high risk contribution panels. The switchgear rooms are provided with area wide incipient detection systems, which were not previously credited in the Fire PRA. A review of the risk benefit of incorporation of the switchgear room area-wide incipient detection system indicated that the resultant risk decrease would be small. Therefore, no credit for the switchgear area-wide incipient detection system is taken. Appendix E describes the approach used for crediting the cable spreading room in-panel incipient detection systems in conjunction with the NUREG-2230 approach.

3.6.4.3 New and Modified Scenarios

New scenarios have been added in the Cable Spreading Room (098), the Switchgear Rooms (067, 068, 070, and 071) and the Control Room (106). A scenario containing all cubicles was added in the A Switchgear rooms (068 and 071) to account for the severe portion of the fire defined as that portion of the cubicle fires that impacts targets up to but excluding cables impacting diesel dynamic loading failures. HEAF scenarios were refined by pulling the HEAF portion of the ignition frequency from the individual cubicle scenarios and merging them into one HEAF scenario for each room. This scenario incorporated the targets associated with the highest risk individual cubicle scenario.

The Cable spreading room added new scenarios. Severe panel scenarios were created which included all targets except the closest tray that contained RCP, Component Cooling Water (CCW) or containment isolation pressure switch cables, which, if damaged, could result in a loss of RCP seal cooling. These trays were excluded and NSPs associated with them were created using the distance to the nearest associated tray. Unit specific NSPs were assigned based on the location of that unit’s cables with respect to high risk ignition sources.

The original Control Room main control board scenarios were based on one scenario with all cabinets impacted. This scenario was split into 4 scenarios that included failure of each combination of adjacent panels.

The table below shows the new scenarios and the basis for their addition to the model. Some of the scenarios were not added but just modified by changing the scenario name.

Table 3-2 – New Scenarios

New Scenarios	Description	Reason for Addition
025-D-HEAF-PTB	480V Load Center 4H-HEAF	Separate HEAF scenario for load center
025-E-HEAF-PTB	480V Load Center 3H-HEAF	Separate HEAF scenario for load center
062-F-1-PTB	3C228A Electrical Cabinets - Severe	Severe scenarios added
062-F-2-PTB	4C228A Electrical Cabinets - Severe	Severe scenarios added
062-J-1-PTB	0C182A Electrical cabinet fire - Severe	Severe scenarios added
062-K-1-PTB	0C182B Electrical cabinet fire - Severe	Severe scenarios added
062-N-PTB	3C810 Electrical cabinet fire - Severe	Severe scenarios added
062-P-PTB	4C810 Electrical cabinet fire - Severe	Severe scenarios added
062-Q-PTB	4C89A/B/C Electrical cabinet fire - Severe	Severe scenarios added
067-HEAF-PTB	HEAF for all cubicle fires (4AB12 bounding Cubicle Fire)	HEAF scenario added for all cubicle fires in PAU
068-AA-PTB	4AA23 Cubicle Fire	Name Change
068-AB-PTB	4AA24 Cubicle Fire	Name Change
068-AC-PTB	XFMR	Name Change
068-AD-PTB	4AA01 through 4AA24 Cubicle Fire	Severe cubicle fire scenario for all cubicles
068-HEAF-PTB	HEAF Fire for all cubicles (4AA01 bounding Cubicle Fire)	HEAF scenario added for all cubicle fires in PAU

Table 3-2 – New Scenarios

New Scenarios	Description	Reason for Addition
070-HEAF-PTB	HEAF fire for all cubicles (3AB22 bounding Cubicle Fire)	HEAF scenario added for all cubicle fires in PAU
071-AB-PTB	3AA01 through 3AA22 Cubicle Fire	Severe cubicle fire scenario for all cubicles
071-HEAF-PTB	HEAF Fire for all Cubicles (3AA21 bounding Cubicle Fire)	HEAF scenario added for all cubicle fires in PAU
093-A-HEAF-PTB	4B01 (4LC A) Load Center-HEAF	Separate HEAF scenario for load center
093-B-HEAF-PTB	4B02 (4LC B) Load Center-HEAF	Separate HEAF scenario for load center
094-A-HEAF-PTB	4B03 (4LC C) Load Center-HEAF	Separate HEAF scenario for load center
094-B-HEAF-PTB	4B04 (4LC D) Load Center-HEAF	Separate HEAF scenario for load center
095-A-HEAF-PTB	3B01 (3LC A) Load Center-HEAF	Separate HEAF scenario for load center
095-B-HEAF-PTB	3B02 (3LC B) Load Center-HEAF	Separate HEAF scenario for load center
096-A-HEAF-PTB	3B03 (3LC C) Load Center-HEAF	Separate HEAF scenario for load center
096-B-HEAF-PTB	3B04 (3LC D) Load Center-HEAF	Separate HEAF scenario for load center
098-AA-XRCPU3U4-PTB	4QR80A/B Fire	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)

Table 3-2 – New Scenarios

New Scenarios	Description	Reason for Addition
098-AK-XRCPU3U4-PTB	3C11/4C11	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-AY-XRCPU3U4-PTB	C-600	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-B-XRCPU3U4-PTB	3/4C260 Relay Panels	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-E-XRCPU3U4-PTB	DDPS RACK	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-F-XRCPU3U4-PTB	3QR80A/B	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-G-XRCPU3U4-PTB	3Q632 through 3QR36	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-P-XRCPU3U4-PTB	3QR37 through 3QR41	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)

Table 3-2 – New Scenarios

New Scenarios	Description	Reason for Addition
098-Q-XRCPU3U4-PTB	4QR32 through 4QR36	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
098-Z-XRCPU3U4-PTB	4QR37 through 4QR41	RCP scenario added for distance to nearest RCP cable tray (Separate NSP value for RCP exclusion for unit tray distance)
101-F-NS-PTB	4D01 125VDC MCC	New Ignition source
106-3C-MCB-1-PTB	MCB Fire impacting 3C01 and 3C02	MCB Scenario separated into adjacent cabinet
106-3C-MCB-2-PTB	MCB Fire impacting 3C03 and 3C04	MCB Scenario separated into adjacent cabinet
106-3C-MCB-3A-PTB	MCB Fire 3C03 and 3C05 (exc 3C05A-D)	MCB Scenario separated into adjacent cubicle up to but excluding RCP cable in adjacent cabinet
106-3C-MCB-3B-PTB	MCB Fire 3C03 and 3C05 (exc 3C03B-H)	MCB Scenario separated into adjacent cubicle up to but excluding RCP cable in adjacent cabinet
106-3C-MCB-4-PTB	MCB Fire impacting 3C05 and 3C06	MCB Scenario separated into adjacent cabinet
106-4C-MCB-1-PTB	MCB Fire impacting 4C01 and 4C02	MCB Scenario separated into adjacent cabinet
106-4C-MCB-2-PTB	MCB Fire impacting 4C03 and 4C04	MCB Scenario separated into adjacent cabinet

Table 3-2 – New Scenarios

New Scenarios	*Description*	*Reason for Addition*
106-4C-MCB-3-PTB	MCB Fire impacting 4C04 and 4C05	MCB Scenario separated into adjacent cabinet
106-4C-MCB-4-PTB	MCB Fire impacting 4C05 and 4C06	MCB Scenario separated into adjacent cabinet

3.6.4.4 Individual Scenario NSP Changes

See FRANX quantification files for updated scenario NSP values associated with new scenarios and NUREG-2230 incorporation.

3.6.4.5 NSP Changes for Hot Gas Layer Scenarios

The following table contains HGL NSP values for the compartment hot gas layer scenarios. These values are from the FMW and NUREG-2180/2230 spreadsheets. NSP for 068-PTB and 071-1-A-PTB are conservative values relative to the FMW results.

Table 3-3 - Fire Zone Hot Gas Layer NSP Values

Scenario	*Source*	*NSP Value*
025-PTB	FMW HGL	3.47E-02
058-PTB	FMW HGL	1.86E-02
061-PTB	FMW HGL	3.67E-02
062-PTB	FMW HGL	1.17E-01
063-PTB	FMW HGL	3.13E-02
067-PTB	FMW HGL	2.48E-01
068-PTB	FMW HGL	3.80E-02

Table 3-3 - Fire Zone Hot Gas Layer NSP Values

Scenario	Source	NSP Value
070-1-A-PTB	FMW HGL	5.46E-02
093-PTB	FMW HGL	1.86E-01
094-PTB	FMW HGL	1.53E-01
095-PTB	FMW HGL	1.48E-01
096-PTB	FMW HGL	2.44E-01
098-PTB	FMW HGL	6.08E-03
098-PTB	FMW HGL	5.43E-03
101-1-A-PTB	FMW HGL	1.45E-01
104-PTB	FMW HGL	2.85E-02
108B-PTB	FMW HGL	1.48E-02
108A-PTB	FMW HGL	3.15E-03

3.6.5 HRA Analysis Update

3.6.5.1 HEP Changes

+ CHFPSTPRCP-F (Failure to stop RCPs given loss of CCW), System Time Window (Tsw) change

  Updated Tsw to reflect the time window for tripping the RCPs with the Framatome PSDS installed, 16 minutes [8].

+ QHFPSTPRCP-F (Failure to stop RCPs given loss of Intake Cooling Water, ICW), no change, RCP trip related; associated with time window for loss of ICW. This timeframe envelopes the timeframe for RCP trip and is therefore not changed.

+ EHFPDROP4KV4A/3A-F (De-energize 4kV Bus 4A following severe fire in the CSR) failed for new RCP seals due to time required exceeding Tsw; not credited in quantification for pre-PSDS model.

- RHFPRCPTRPBC-F (Trip RCP B/C for a fire induced spurious start following reactor trip), failed for new RCP seals due to time required exceeding Tsw, not credited in quantification for pre-PSDS model
- AHFPSGLVL-F (Control SG level at the ASP to maintain secondary heat sink, using Alternate Shutdown Panel, ASP, wide range instrumentation), updated calc of Tsw based on including S/G steaming in time to overfill calculation. HEP refinement to reduce conservatism in HEP.
- FTISEALCLG-LOCAL (fail to isolate seal cooling prior to spurious initiation of seal cooling subsequent to initial loss of cooling), new action, 0.01 screening value, detailed HEP to be developed in conjunction with review and update of associated procedures for RCP seal replacement. System Time Window (Tsw) is 59 minutes [4].
- MHFPRWST358-F (Failure to establish alternate suction path from RWST to the charging pumps), updated Tsw to 16 minutes, resulted in failure of operator action. This HEP was not a significant contributor to baseline, pre-RCP seal replacement model.
- RHFPRESET-F (Failure to reset SI signal to allow seal injection via charging to be restored), updated Tsw to 16 minutes [8]. This HEP was not a significant contributor to baseline, pre-RCP seal replacement model.

3.6.5.2 Updated Dependency Evaluation

The following t delay (Td) override was incorporated into the HRA during cutset reviews with NEE.

Table 3-4 – Time Delay Adjustments

HFE	*Conditional HFE*	*Td Override (Min)*	*Comment*
GHFBLFEEDL-F	AFPAFWTHROT-F	52	The Td for GHFBLFEEDL-F/AFPAFWTHROT-F should be increased to 52 mins. This reflects the 15 mins after failure of AFPAFWTHROT (15+37)

3.7 RESULTS

3.7.1 Risk Analysis Results

3.7.1.1 Summary Tables and Figures

The table below shows current risk results with adjusted values. Adjusted results incorporate CCDP/CLERP adjustments for control room abandonment scenarios using the same methodology applied in the NFPA 805 LAR model. Adjusted results also included setting RCP loop specific asymmetric cooling (ZZASYCOOLA/B/C) flags to true to eliminate non minimal cutsets.

Table 3-5 – Risk Results Unit 3 and Unit 4 Fire CDF and LERF

Risk Metric	Truncation	Base Results	# of cutsets	Adjusted results
U3 CDF	1.00E-10	7.23E-05	44944	7.99E-05
U3 LERF	1.00E-12	1.82E-06	86128	1.86E-06
U4 CDF	1.00E-10	7.73E-05	15482	7.80E-05
U4 LERF	1.00E-12	1.82E-06	83633	1.88E-06

Table 3-6 – FRANX Database Files

Risk Metric	FRANX Database	File Size	Top Gate	Truncation
U3 CDF	PTNRev14_RCPSealMod_Unit3_CDF_{Base Calc}	122,828 KB (08/09/22, 7:29am)	ALLTOPS	1.00E-10
U3 LERF	PTNRev14_RCPSealMod_Unit3_LERF_{Base Calc}	157,696 KB (08/09/22, 7:08am)	LERFOLRM1	1.00E-12
U4 CDF	PTNRev14_RCPSealMod_Unit4_CDF_{Base Calc}	121,956 KB (08/09/22, 7:37am)	U4ALLTOPS	1.00E-10
U4 LERF	PTNRev14_RCPSealMod_Unit4_LERF_{Base Calc}	122,356 KB (08/09/22, 7:11am)	U4LERFOLRM1	1.00E-12

To run the FRANX model the databases from Table 3-6 and the following model files were used:

- Fault Tree: ptnrev14F_RCPSealMod_8.8.22.caf
- Recovery File: ptnrev14FHFEAll_RCPSealMod_8.6.22.recv
- RR Database: ptnrev14FHFEAll_RCPSealMod_8.8.22.rr
- Flag File: ptnrev14fire.flg
- MUTEX File: MUTEX.cut

The figures below identify the zones with highest risk contribution for Unit 3 and Unit 4 Fire CDF and LERF results.

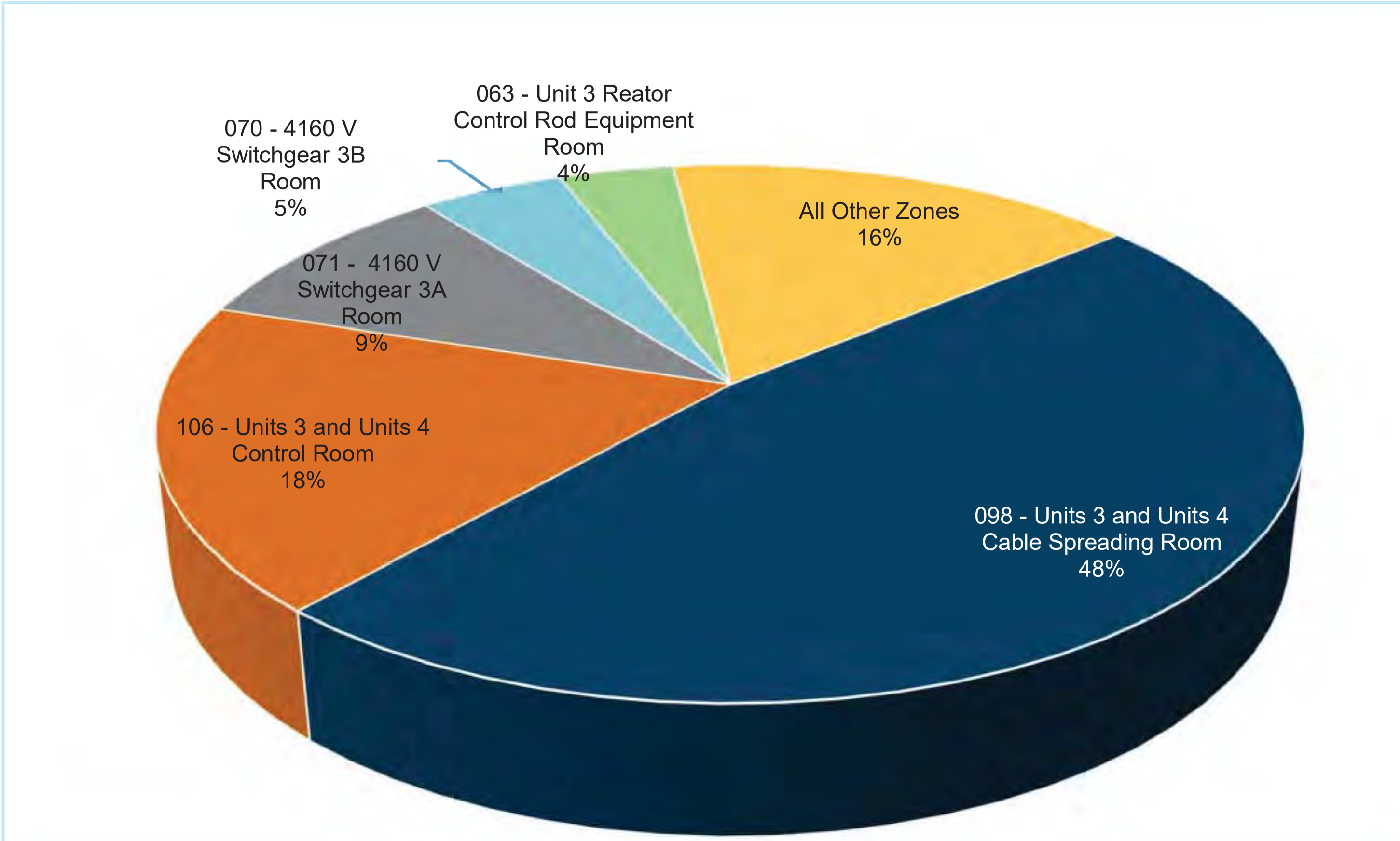


Figure 3-2 Risk by Fire Zone (Unit 3 CDF)

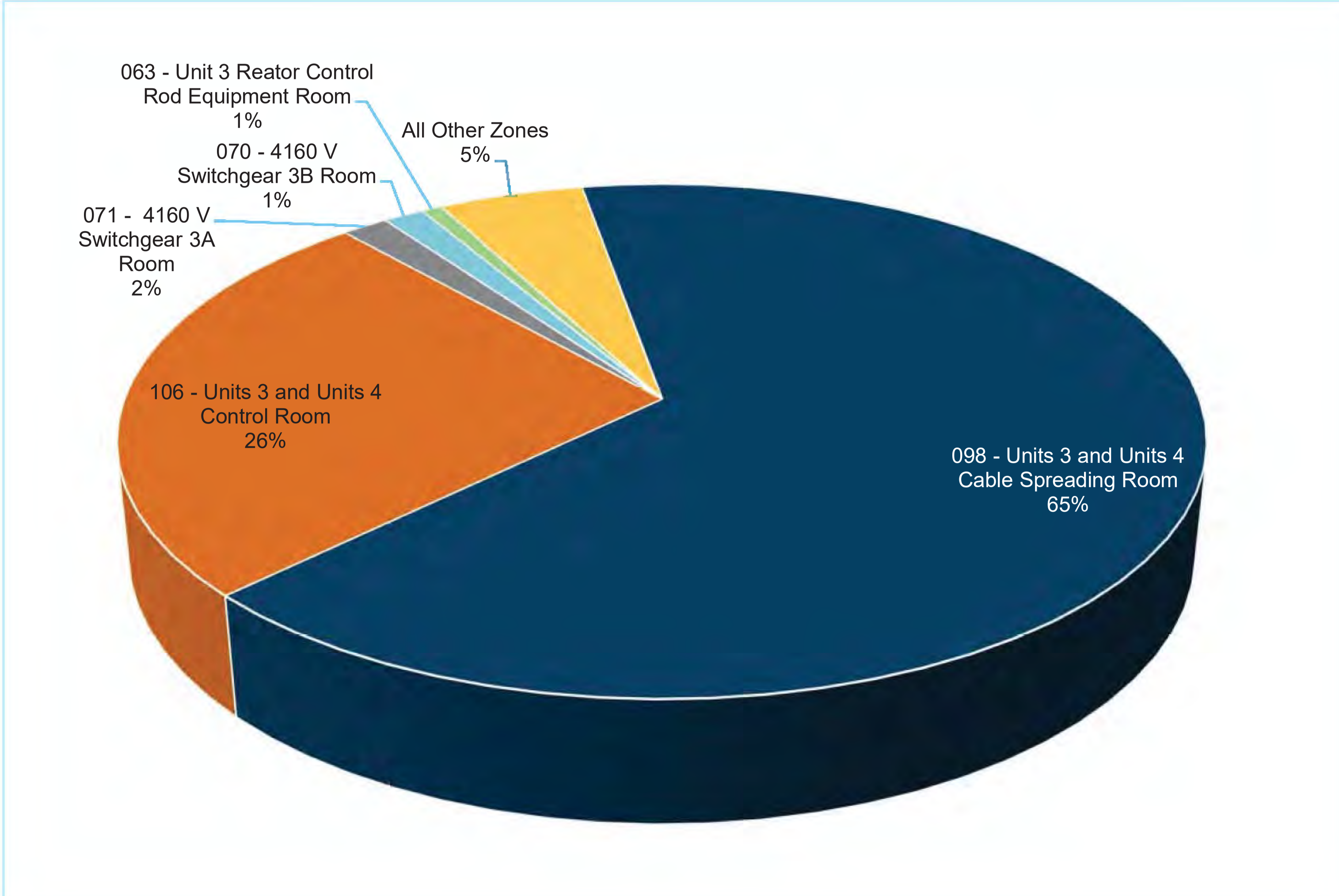


Figure 3-3 Risk by Fire Zone (Unit 3 LERF)

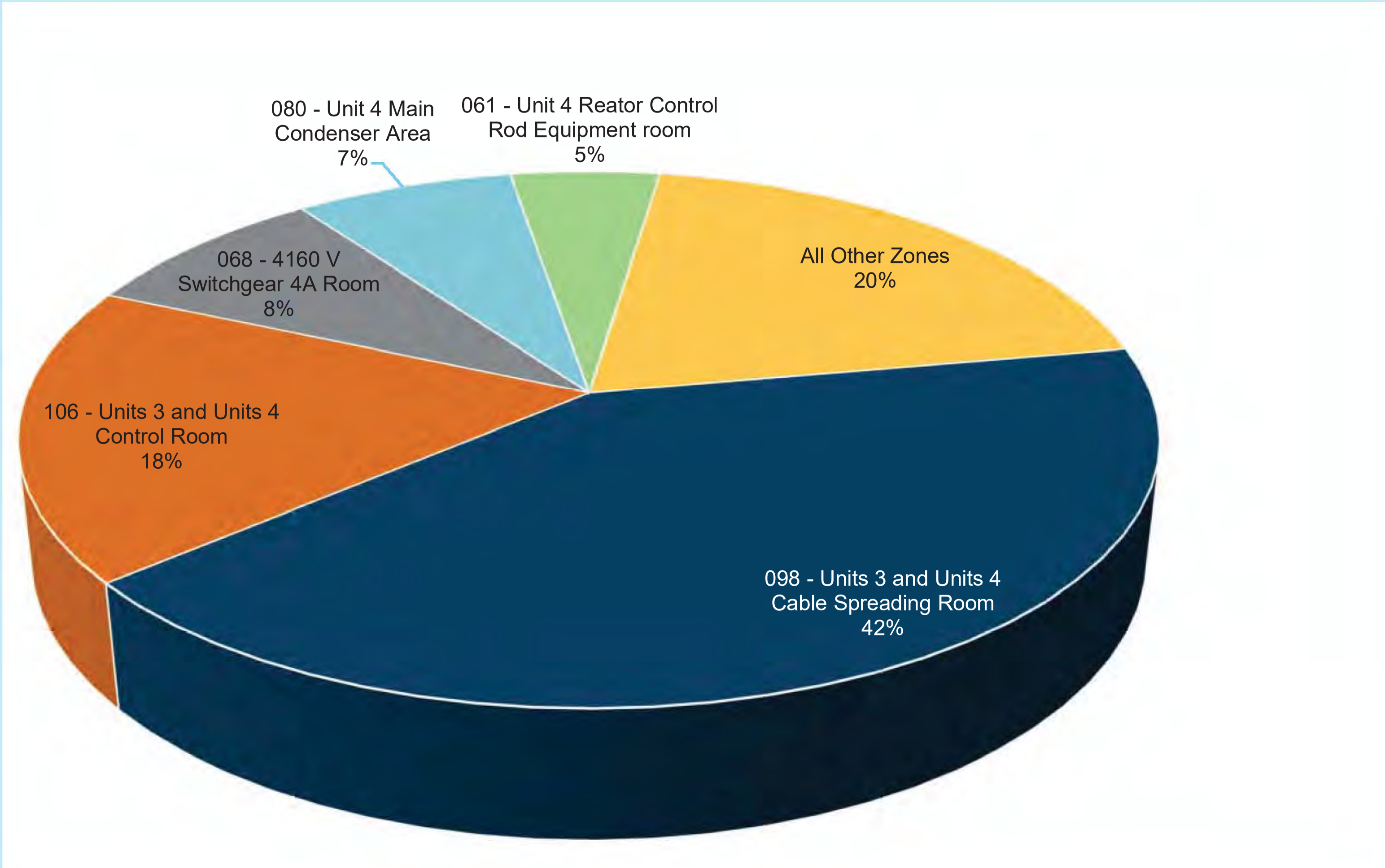


Figure 3-4 Risk by Fire Zone (Unit 4 CDF)

Zone 080 contribution is associated with conservative scenario definition. Refinement of these scenarios will be incorporated in future Fire PRA model updates.

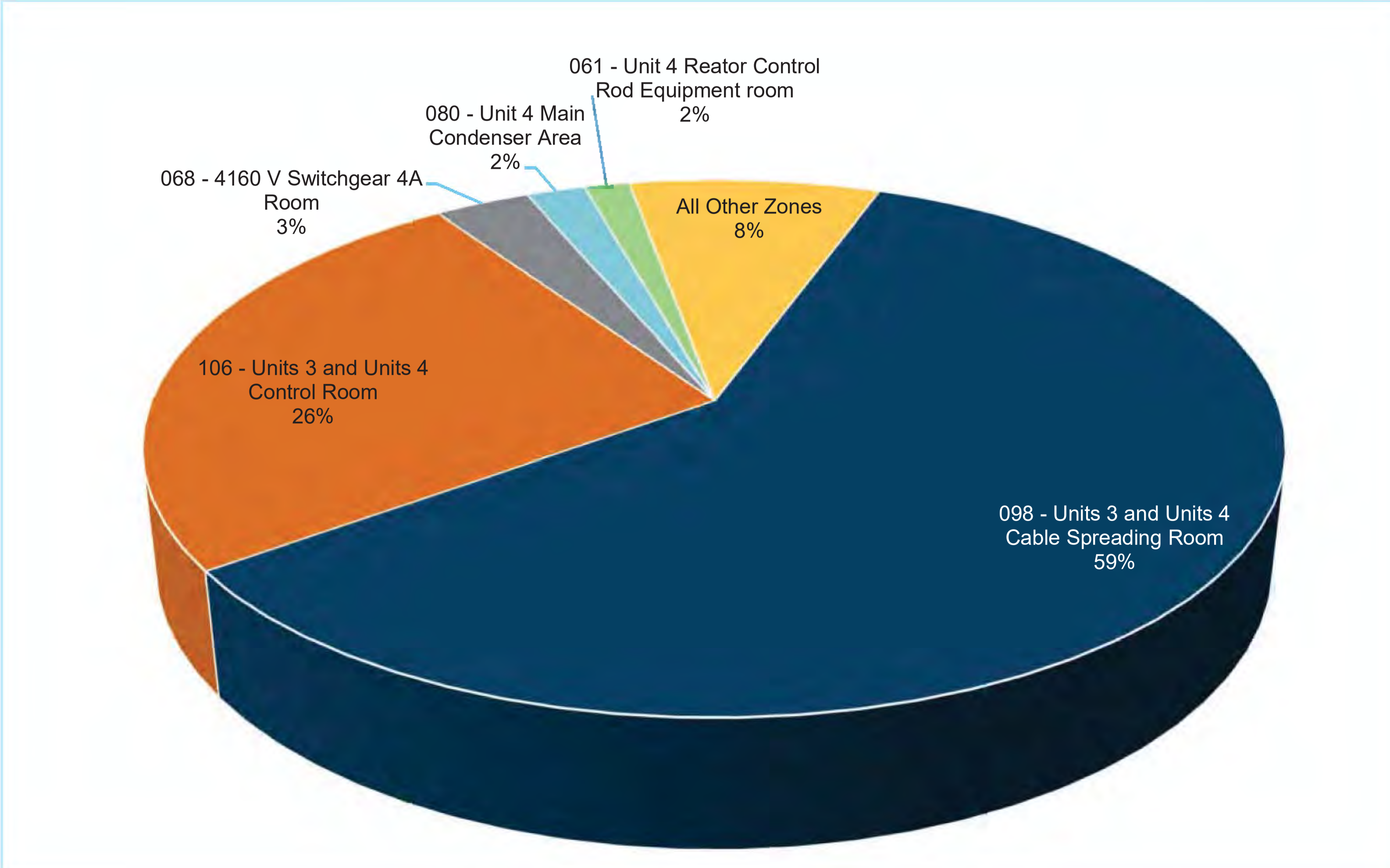


Figure 3-5 Risk by Fire Zone (Unit 4 LERF)

Zone 080 contribution is associated with conservative scenario definition. Refinement of these scenarios will be incorporated in future Fire PRA model updates.

3.7.2 Delta Risk Results

3.7.2.1 Quantification of Baseline Risk, With Previous RCP Seal Modeling

+ Set RCP seal failure probability BE, ZZRCPSL3/4A/B/C, to value in previous model (adjusted for per RCP application in current model).
+ Set asymmetric cooling flag, ZZASYCOOLA/B/C, to 0
+ Set RCP seal leakoff isolation valve hot short probability to 0, to eliminate seal failure mode with RCP seal leakoff isolation, CV-3/4-303A/B/CFTRO_1 set to 0
+ Set operator action FTISEALCLGLOCAL, for isolation of seal cooling to prevent seal failure due to thermal shock, to 0

3.7.2.2 Delta Risk Results Summary

Table 3-7 – Fire Delta Risk

	CDF post mod	*CDF pre mod, with refinements**	*Delta CDF*	*% Increase*	*LERF post mod*	*LERF pre mod, with refinements**	*Delta LERF*	*% Increase*
U3	7.99E-05	7.47E-05	5.20E-06	7%	1.86E-06	1.66E-06	2.00E-07	12%
U4	7.80E-05	7.37E-05	4.30E-06	6%	1.88E-06	1.70E-06	1.80E-07	11%

* - CDF/LERF pre-mod with refinements results specified above incorporate the same Fire PRA modeling refinements implemented in the post mod model into the pre-mod model to ensure that the delta risk specified is based on a consistent level of Fire PRA model refinement

3.7.3 Cutset Review

See project correspondence file for a summary of discussions, action items and their resolution associated with the cutset review performed for the RCP seal replacement Fire PRA model.

3.7.4 Quantification Software

The following software was used for Fire PRA model quantification:

+ CAFTA 6.0b
+ FRANX 4.4
+ UNCERT 4.0
+ FTREX 1.8 and 2.0

4.0 References

1. NUREG-2230: Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinet Fires in Nuclear Power Plants. EPRI, Palo Alto, CA, and the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Washington, D.C.: 2020. 3002016051/NUREG-2230.
2. NUREG-2233: Methodology for Modeling Transient Fires in Nuclear Power Plant Fire Probabilistic Risk Assessment, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Washington, D.C. and Electric Power Research Institute (EPRI), Palo Alto, CA: 2019. NUREG-2233 and EPRI 3002016054.
3. NUREG-2178, V2: Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire, Volume 2: Fire Modeling Guidance for Electrical Cabinets, Electric Motors, Indoor Dry Transformers, and the Main Control Board, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA: 2019. NUREG-2178, V2, and EPRI 3002016052.
4. Framatome Calculation, 32-9349361-000, Turkey Point RCP Seal – PSDS Cooldown Calculation, Revision 0, dated 6/3/22 (JH Correspondence Log Item 0F6S001-EML-044)
5. Framatome Document Number 38-9349703-000 (JH Correspondence Log Item (0F6S001-EML-041)
6. Passive Shutdown Seal – Evaluation of Failure to Actuate for Framatome RCP Seals, Framatome Engineering Information Record, Document No. 51-9351505-000, Revision 0, dated 08/12/2022
7. Passive Shutdown Seal – Evaluation of Failure to Remain Sealed, Framatome Engineering Information Record, Document No. 51-9348566-001, Revision 1, dated 08/15/2022
8. Passive Shutdown Seal – PRA Evaluation of Spurious Actuation, Framatome Engineering Information Record, Doument No. 51-9227814-004, Revision 3, dated 08/15/2022
9. NUREG-2180: Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE). U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD. Final report, December 2016.
10. PTN-BFJR-00-001, "Turkey Point PRA Model Update," Revision 14, October 2021.

APPENDICES TABLE OF CONTENTS

Appendix A Full Power Internal Events (FPIE)/Internal Flood (IF) Impact of Seal Modification

The criteria for NFPA 805 post transition fire protection program changes requiring NRC review is specific to the Fire PRA results. However, to estimate the impact on total plant risk, the FPIE/IF risk numbers are reviewed with respect to the potential impact of the seal modification. A conservative estimate of the impact of the seal modification on the internal events and flooding models can be made by assuming that the delta risk for the Fire PRA as a fraction of total fire risk can be applied to the FPIE/IF risk values. This estimate is conservative since the FPIE/IF models are not as sensitive to seal failure given that the scenarios for which the RCP seal would actuate are limited to blackout scenarios and random failures of the thermal barrier cooling and seal injection systems which are far less likely than fire induced failures of these systems.

The current model of record FPIE/IF risk for PTN is (PTN-BFJR-00-001, Revision 14, [10]):

Unit	CDF	LERF
Unit 3	1.56E-07	3.66E-09
Unit 4	1.55E-07	3.62E-09

Fire PRA Total Risk post Framatome PSDS installation and the risk using the Fire PRA with refinements incorporated for the current RCP seals and associated Delta Risk (per reactor year) are: (from Section 3.7.2.2)

Fire Risk

Unit	CDF post mod	CDF pre mod, with refinements	ΔCDF	% Increase	LERF post mod	LERF pre mod, with refinements	ΔLERF	% Increase
Unit 3	7.99E-05	7.47E-05	5.20E-06	7%	1.86E-06	1.66E-06	2.00E-07	12%
Unit 4	7.80E-05	7.37E-05	4.30E-06	6%	1.88E-06	1.70E-06	1.80E-07	11%

Application of the above % increase to the FPIE/IF model results reported above results in the following post mod and delta risk values (per reactor year).

FPIE/IF Risk

Unit	CDF post mod	CDF pre mod, with refinements	ΔCDF	% Increase	LERF post mod	LERF pre mod, with refinements	ΔLERF	% Increase
Unit 3	1.67E-07	1.56E-07	1.07E-08	7%	4.10E-09	3.66E-09	4.41E-10	12%
Unit 4	1.64E-07	1.55E-07	8.83E-09	6%	4.00E-09	3.62E-09	3.83E-10	11%

The total mod risk including the conservative estimate of the FPIE/IF risk increase is:

Total Fire and FPIE/IF Risk

Unit	CDF post mod	CDF pre mod, with refinements	ΔCDF	% Increase	LERF post mod	LERF pre mod, with refinements	ΔLERF	% Increase
Unit 3	8.01E-05	7.49E-05	5.21E-06	7%	1.86E-06	1.66E-06	2.00E-07	12%
Unit 4	7.82E-05	7.39E-05	4.31E-06	6%	1.88E-06	1.70E-06	1.80E-07	11%

Appendix B PRA Quality (Peer Reviews, F&O Closures, Open F&O Impact)

The FPIE PRA model of record for this evaluation is Revision 14, as documented in PTN-BFJR-00-001, Revision 14, "Turkey Point PRA Model Update," (Reference 1).

The *PRA models have been assessed against RG 1.200, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 2. The Internal Events PRA model was subject to a self-assessment and a full-scope peer review conducted in January 2002. In April 2011, a focused peer review was performed assessing the human reliability analysis (HRA) and internal flooding analysis portions of the PRA against the 2009 Standard's requirements. A focused peer review was performed in October 2013, to assess portions of the PRA model associated with common-cause failure analysis, Level 2 analysis, and interfacing system LOCAs.*

The Internal Events PRA technical adequacy has previously been reviewed by the NRC in previous applications for transition to NFPA-805 and relocation of surveillance frequency requirements to licensee control. No PRA upgrades as defined by the ASME PRA Standard RA-Sa-2009 have occurred to the Internal Events PRA model.

The Fire PRA model was subject to a self-assessment and a full-scope peer review conducted in February 2010. A subsequent peer review, performed in March 2012, was a focused scope peer review addressing the FSS, HRA, and PRM technical elements of the Fire PRA.

Finding closure reviews were issued on the identified PRA models in February and June of 2019. Open findings were reviewed and closed using the process documented in Appendix X to NEI 05-04, NEI 07-12 and NEI 12-13, "Close-out of Facts and Observations" (F&Os) as accepted by NRC in the staff memorandum dated May 3, 2017 (ML17079A427). The results of this review have been documented and are available for NRC audit.

Table B-1 provides a summary of the remaining findings and open items.

The only remaining open issues from the peer review process for Turkey Point are associated with the Fire PRA. The Table below provides a listing of the open findings and the status of their resolution as well as an assessment of the impact on this evaluation.

Finally, the open items in the PTN Change Database were examined for their potential impact on the risk analysis for this LAR. The open items involved the removal of conservatisms or documentation changes. The former would only reduce the estimated impact of the RCP Seal modification; the latter would have no effect. The remaining few were reviewed and judged to have minimal impact on the risk analysis of this LAR.

Table B-1: Turkey Point Open Peer Review Issues

Supporting Requirement	Issue	Evaluation
UNC-A1-01	An uncertainty analysis was not performed for the current fire PRA. The previous documented uncertainty analysis is multiple revisions old and has not been maintained with the multiple changes implemented in the fire PRA. The uncertainty and sensitivity calculation(s) need to be performed for the latest FPRA model.	The uncertainty and sensitivity analysis has since been updated and has been revised to incorporate the impact of the RCP seal replacement. See Appendix C.
FSS-C1-02	In reviewing the calculation of the NSP development for the hot gas layer scenario in PAU 068, it was identified that the equation is incorrect and not appropriately accounting for the transient frequency. The use of this equation and its inputs should be reviewed for all scenarios of this type to ensure that the applicable inputs are being used correctly in the overall calculation. This is a concern for those PAUs that model full room burnout transient scenarios. The transient frequency is only included in the hot gas layer scenario.	This issues impact has been reviewed and determined to be non-risk significant. The current model has been updated to correct the discrepancy noted.
FSS-C1-03	The NSP calculation for TGO is calculated improperly for scenario 078-J-PTB. The NSP calculation uses a motor as the HRR. The ignition source is oil for this source. The NSP for this scenario needs to be updated to appropriately account for the ignition source type and characteristics. Other TGO fires should be reviewed for this same incorrect calculation.	This issue's impact has been reviewed and determined to be non-risk significant. The current model has been updated to correct the discrepancy noted.

REFERENCES cited in above text:

REFERENCES

1. PTN-BFJR-00-001, "Turkey Point PRA Model Update," Revision 14, October 2021.

Appendix C Other LAR Inputs

C.1 UNCERTAINTY ANALYSIS

C.1.1 Convergence Review

The convergence review summarized below is conservatively based on the FRANX results prior to adjustment for control room abandonment scenario CCDPs and CLERPs. Inclusion of the adjustment will add the same increase in risk for the different truncation quantifications.

Table C-1: Quantification Convergence Summary

	Risk Results at Specified Truncation Values					Risk Increase at Specified Truncation Values			
	E-9	E-10	E-11	E-12	E-13	E-9 to E-10	E-10 to E-11	E-11 to E-12	E-12 to E-13
U3 CDF	6.80E-05	7.22E-05	7.41E-05			6.2%	2.6%		
U3 LERF			2.38E-06	2.61E-06	2.72E-06			9.7%	4.2%
U4 CDF	7.24E-05	7.72E-05	7.90E-05			6.6%	2.3%		
U4 LERF			2.26E-06	2.48E-06	2.57E-06			9.7%	3.6%

The results used for this application are based on 1E-10 truncation for CDF and 1E-12 truncation for LERF.

C.1.2 Sensitivities – UNL and RCP Seal Failure Value

Sensitivity for UNL components not failed (quantified for CDF only, at E-9 truncation, without CCDP adjustments for control room abandonment scenarios)

Table C-2: UNL Sensitivity

Units	*Baseline*	*no UNL failures*	*% Decrease*
Unit 3 CDF	7.22E-05	5.67E-05	21%
Unit 4 CDF	7.72E-05	6.15E-05	20%

The decrease in risk noted above is based on a non-conservative assumption that all UNL components are unaffected by any fire scenarios. Therefore, the actual reduction in risk should the UNL components be credited is expected to be significantly lower than the value specified above.

C.1.3 Sensitivity of Fire Risk Results to RCP Seal Failure Probability

Table C-3: 2 X RCP Failure Sensitivity

	Baseline		*2 X RCP Seal Failure Probability (3.32E-03)*			
Units	*CDF*	*LERF*	*CDF*	*% Increase*	*LERF*	*% Increase*
Unit 3	8.01E-05	1.86E-06	8.40E-05	5.0%	1.89E-06	1.4%
Unit 4	7.82E-05	1.88E-06	8.23E-05	5.4%	1.91E-06	1.4%

Table C-3: 0.5 X RCP Failure Sensitivity

	Baseline		0.5 X RCP Seal Failure Probability (8.3E-04)			
Units	*CDF*	*LERF*	*CDF*	*% Decrease*	*LERF*	*% Decrease*
Unit 3	8.01E-05	1.86E-06	7.79E-05	2.8%	1.85E-06	0.5%
Unit 4	7.82E-05	1.88E-06	7.58E-05	3.1%	1.86E-06	1.1%

C.1.4 Uncertainty Matrix

NUREG/CR-6850 is broken into 16 distinct tasks. The uncertainty contribution to the analysis for each of these tasks is outlined in Table C-4 below.

Table C-4: Uncertainty Matrix

Task No.	*Sources of Uncertainty*	*Sensitivity of the Results to the Source(s) of Uncertainty.*
1	This task poses a limited source of uncertainty beyond the credit taken for boundaries and partitions.	During scenario development, the zone of influence was not limited to the physical analysis unit boundary. If the zone of influence included targets in adjacent fire zones, these targets were also included, regardless of their fire zone location. In addition, the multi-compartment analysis further reduces this uncertainty by addressing the potential impact of failure of partition elements on quantification. This source of uncertainty is not impacted by the RCP seal replacement.

Table C-4: Uncertainty Matrix

Task No.	*Sources of Uncertainty*	*Sensitivity of the Results to the Source(s) of Uncertainty.*
2	This task poses perhaps the highest potential for error if not uncertainty. The mapping of basic events to components requires not only the consideration of failure modes (active versus passive) but an understanding of the Appendix R functions not previously considered risk significant in the FPIE model. When performed correctly, the only uncertainty is related to the MSO process.	The potential for uncertainty is reduced as a result of multiple overlapping tasks including the MSO expert panel. Additional internal reviews and the change evaluation process performed in support of the NFPA 805 LAR application further reduce uncertainty in this task. No additional sources of uncertainty are introduced by the RCP seal replacement since no new components requiring mapping are added to the model.
3	No treatment of uncertainty is typically required for this task beyond the understanding of the cable selection approach (i.e., mapping an active basic event to a passive component for which power cables were not selected). Additionally, PRA credited components for which cable routing information was not provided represent a source of uncertainty (conservatism) in that Y3 components could be assumed failed unnecessarily	The limited number of Y3 components (most active components credited in the fire PRA were included in the Safe Shutdown Analysis data) as well as the crediting by exclusion of Y3 components (where justified) helps to reduce unnecessary conservatism. Sensitivity quantifications were performed in which the Y3 components were assumed to be available (as opposed to damaged) for all fire scenarios. The results of these sensitivity runs are provided in Section C.1.2. The actual configuration in which the Y3 components are lost in some fire zones but not all fire zones would result in a smaller reduction in CDF/LERF than that identified in the sensitivity evaluation. Therefore, the impact of this uncertainty is not considered particularly significant in light of credit for these components by exclusion where their loss was creating a significant impact on the risk of a given fire scenario and where their availability could be substantiated by general plant design/layout. The RCP seal replacement does not introduce any new Y3 components into the analysis.

Table C-4: Uncertainty Matrix

Task No.	*Sources of Uncertainty*	*Sensitivity of the Results to the Source(s) of Uncertainty.*
4	Qualitative screening was not performed; however, structures were eliminated from the global analysis boundary and ignition sources deemed to have no impact on the fire PRA were excluded from the quantification based on qualitative screening criteria. The only criterion subject to uncertainty is the potential for plant trip	In the event that a structure which could lead to a plant trip was excluded incorrectly, its contribution to CDF would be small (with a CCDP commensurate with base risk) and would likely be offset by inclusion of the additional ignition sources on the reduction of other scenario frequencies. A similar argument can be made for ignition sources for which scenario development was deemed unnecessary. This source of uncertainty is not altered by the RCP seal replacement.
5	A reactor trip is assumed as the initiating event for all quantification. This is somewhat conservative since not all fires postulated will result in a plant trip.	FPIE and fire PRA peer reviews (including the F&O resolution process), internal assessments, and the NFPA 805 LAR change evaluation process are useful in exercising the model and identifying weaknesses with respect to this assumption. No changes to the assumed probability that a reactor trip occurs in conjunction with a fire for the RCP seal replacement.
6	Ignition source counting is an area with inherent uncertainty; however, the results are not particularly sensitive to changes in ignition source counts. The primary source of uncertainty for this task is associated with the frequency values from NUREG/CR-2169 [6] which result in uncertainty due to variability among plants along with some significant conservatism in defining the frequencies, and their associated heat release rates, based on limited fire events and fire test data.	The conservatism in the ignition frequency data, which is also linked to conservatism in non- suppression probability data specified in NUREG-2169 [6] appears to introduce a significant conservatism. This uncertainty/conservatism is not altered by the RCP seal replacement.

Table C-4: Uncertainty Matrix

Task No.	Sources of Uncertainty	Sensitivity of the Results to the Source(s) of Uncertainty.
7	Other than screening out potentially risk significant scenarios (ignition sources), there is no uncertainty from this task on the fire PRA results.	Quantitative screening is limited to refraining from further scenario refinement of those scenarios with a resulting CDF/LERF below the screening threshold. All of the results were retained in the cumulative CDF/LERF. Therefore this task is not a source of uncertainty in the Fire PRA.
8	The approach taken for this task included: 1) the use of generic fire modeling treatments in lieu of conservative scoping analysis techniques and 2) limited detailed fire modeling was performed to refine the scenarios developed using the generic fire modeling solutions. The primary conservatism introduced by this task is associated with the heat release rates specified in NUREG/CR 6850 [1] and NUREG-2178, Volume 1 [7]	The employment of generic fire modeling solutions did not introduce any significant conservatism. Detailed fire modeling was only applied where the reduction in conservatism was likely to have a measurable impact. Detailed fire modeling was performed under Task 11 where appropriate including the application of multi-point treatments based on split fractions for fires impacting only the ignition source versus fires impacting external targets. The NUREG/CR 6850 [1] and NUREG-2178, Volume 1 [7] heat release rates introduce significant conservatism given the limited fire test data available to define the heat release rates and the associated fire development timeline. Some additional scenario refinement was applied to more realistically define the risk associated with the RCP seal replacement modification.
9	Uncertainty considerations are limited to errors in circuit failure analysis where a cable was deemed incapable of causing loss of a particular function credited in the fire PRA. Similar to Task 2 (with the exception of the MSO process), this task has no associated uncertainty when performed correctly.	Circuit analysis was performed as part of the Appendix R Analysis. Refinements in the application of the circuit analysis results to the fire PRA were performed on a case by case basis where the scenario risk quantification was large enough to warrant further analysis. Therefore, the uncertainty/conservatism which remains in the evaluation is associated with scenarios which do not contribute significantly to the overall fire risk. No new circuit analysis was performed in support of the RCP seal replacement Fire PRA update.

Table C-4: Uncertainty Matrix

Task No.	*Sources of Uncertainty*	*Sensitivity of the Results to the Source(s) of Uncertainty.*
10	The uncertainty associated with the applied conditional failure probabilities poses competing considerations. On the one hand, a failure probability for spurious operation could be applied based solely on cable scope without consideration of less direct fire affects (e.g., a failure likelihood applied to the spurious operation of an MOV without consideration of the fire-induced generation of spurious signal to close or open the MOV). On the other hand, a failure probability for spurious operation could be applied despite the absence of cables capable of causing spurious operation in a given location	Circuit failure mode likelihood analysis was generally limited to those components where spurious operation could not be caused by the generation of a spurious signal. This approach limited the introduction of non-conservative uncertainties. For the 'simple' cases, the potential exists for assuming a failure likelihood greater than 0 versus 0 (or random) failure likelihood in some areas where the cables capable of causing spurious operation are not located. Additional refinement to this approach was performed, as necessary, on risk significant scenarios. So the application of further circuit failure probabilities is considered to have minimal impact on the results. The use of NUREG/CR-7150, Volume 2 [8] Circuit Failure Mode and Likelihood probabilities ensured that the latest state of knowledge related to the likelihood of a particular failure mode is addressed in the analysis. The primary uncertainty would be in limiting the application of the associated failure likelihoods to specific components of concern and not all cables where applicable and the conservatism associated with the values used and the assumption that fire damage to a cable within a zone of influence results in the cable failing (1.0 probability) or failing at the hot short probability associated with the circuit failure mode. A new hot short probability was incorporated for the DC, fail open, RCP No. 1 seal leakoff valve isolation valves; using the appropriate value specified in NUREG-7150, Volume 2.

Table C-4: Uncertainty Matrix

Task No.	Sources of Uncertainty	Sensitivity of the Results to the Source(s) of Uncertainty.
11	The primary uncertainty in this task is in the area of target failure probabilities. Conservative heat release rates may result in additional target damage. Non-conservative heat release rates would have an opposite effect. Credit for fire brigade response and detection are based on NUREG-2169 [6] data as well as incorporation of interruptible fire approaches using NUREG-2230 guidance.	Detailed fire modeling was performed only on those scenarios which otherwise would have been notable risk contributors and only where removal of conservatism in the generic fire modeling solution was likely to provide benefit either via a smaller zone of influence or to credit automatic or manual suppression. Fire modeling was used to evaluate the time to abandonment for control room fire scenarios for a range of fire heat release rates. The analysis methodology conservatism is primarily associated with conservatism in the heat release rates specified in NUREG/CR 6850 [1] and NUREG-2178, Volume 1 [7]. Some additional scenario refinement was applied to more realistically define the risk associated with the RCP seal replacement modification.
12	Human error probabilities represent a potentially large uncertainty for the fire PRA given the importance of human actions in the base model. Since many of the HEP values were adjusted for fire, the joint dependency multipliers developed for the FPIE model also represent a potential for introducing a degree of conservatism.	Conservative HEP adjustments were made to the nominal HEP values used in the FPIE model per the guidance in NUREG-1921 [9] methodology. A Detailed analysis was performed for all fire specific HFEs. A floor value of 1E-05 was applied for all combinations.
13	Since this is a qualitative evaluation, there is no quantitative impact with respect to the uncertainty of this task.	Seismic fire interaction has no impact on fire risk quantification.

Table C-4: Uncertainty Matrix

Task No.	*Sources of Uncertainty*	*Sensitivity of the Results to the Source(s) of Uncertainty.*
14	As the culmination of other tasks, most of the uncertainty associated with quantification has already been addressed. The other source of uncertainty is the selection of the truncation limit	A sensitivity evaluation of the truncation limit used in the analysis is provided in Section C.1.1.
15	This task does not introduce any new uncertainties but is intended to address how uncertainties may impact the fire risk.	N/A
16	This task does not introduce any new uncertainties to the fire risk.	The documentation task compiles the results of the other tasks. See specific technical tasks for a discussion of their associated uncertainty and sensitivity

A discussion of the combined fire PRA specific HLRs and SRs, per the ANS/ASME standard [4], related to uncertainty are provided below:

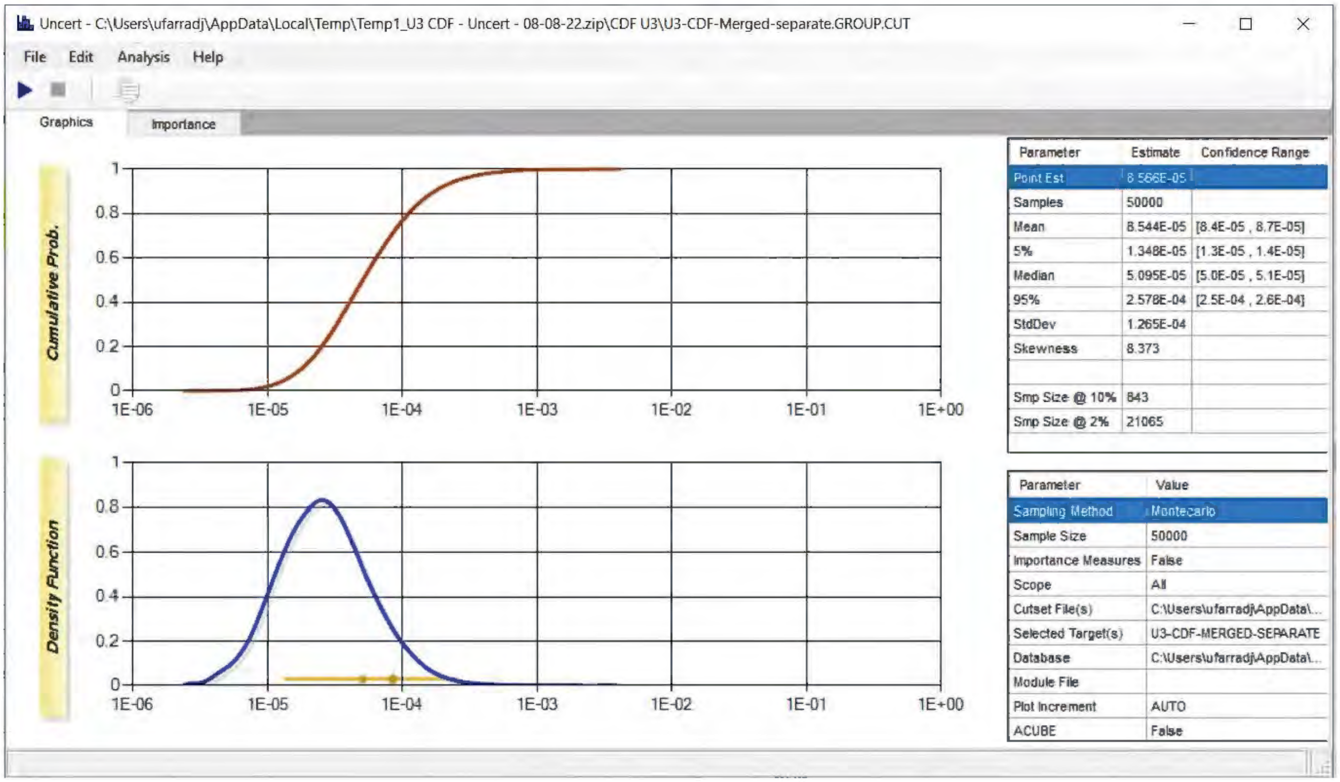
+ PRM-A4: Uncertainties associated with location of equipment and cables are associated with unknown location components and their exclusion via assumed routing. This is addressed by the requirements of FSS-E4 with respect to impact on fire scenario development, discussed below.
+ FQ-F1: References requirements of HLR-QU-F and LE-G (with QU-F4 and LE-G4 specifically related to characterization of model uncertainty and assumptions). These SRs are addressed by this report in Table C.2-1 above as well as the parametric uncertainty analysis provided in Section C.2.2. Convergence/truncation evaluations (addressing QU-B3) are addressed in Section C.1.1.
+ HLR-FSS-E, FSS-H5, FSS-H9: Uncertainties associated with fire modeling have been addressed in the Fire Modeling Analyses to ensure that the use of fire modeling correlations is consistent with the guidance and limitations specified in NUREG-1824, Supplement 1.
+ FSS –E4: Assumed cable routing (exclusion of Y3 components) has been performed based on an evaluation of the routing of required cables for the associated components with respect to the location in which they are to be excluded. Therefore, no uncertainty is associated with this activity.
+ IGN-A10, B-5: Fire Ignition Frequency calculation is consistent with NUREG-2169 [6] frequencies. Frequencies were conservatively assumed to not need a Bayesian update from the industry frequencies. An update for current plant data would be expected to reduce the total plant bin frequencies.

- CF-A2: Used NUREG-7150, Volume 2 [8] as the basis for failure likelihood and hot short duration factors.
- HLR-UNC-A: See tabulation of uncertainties by NUREG/CR-6850 tasks is provided above.

C.1.5 Parametric Uncertainty

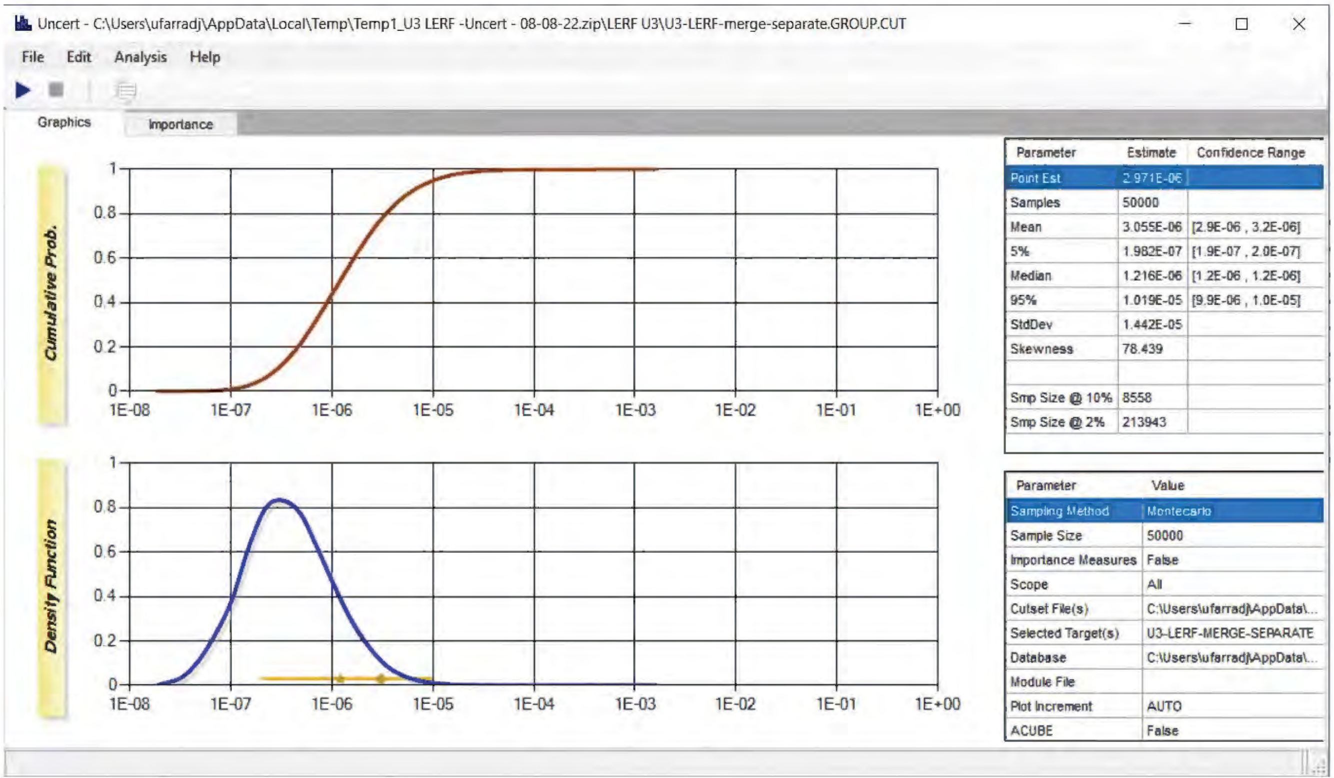
Parametric uncertainty has been performed using unadjusted CCDP/CLERP and unfactored CAFTA cutsets (to allow evaluation of uncertainty associated with ignition frequency, severity factor and non-suppression probability separately). The results of the UNCERT model quantification are provided below. These results show good correlation between the UNCERT calculated mean and the point value risk quantifications.

U3 CDF



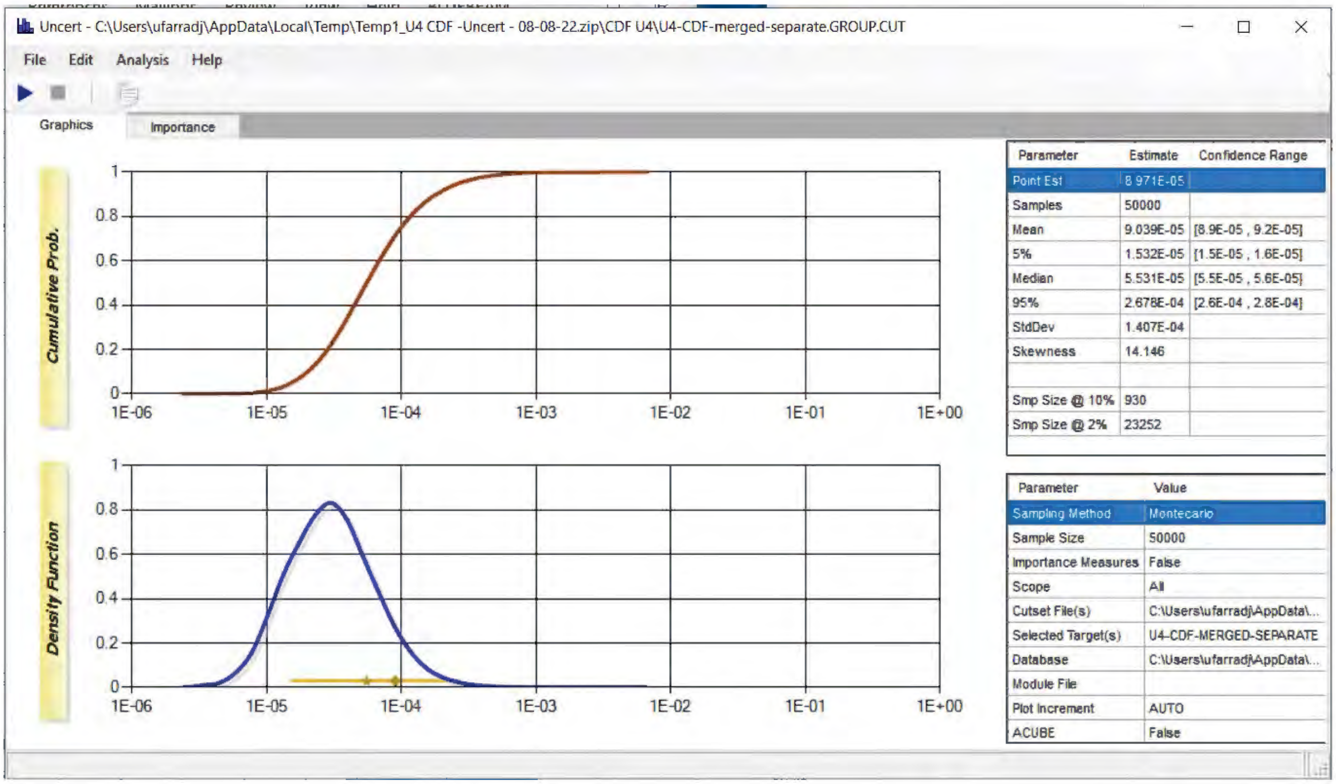
Uncert - C:\Users\ufarradj\AppData\Local\Temp\Temp1_U3 CDF - Uncert - 08-08-22.zip\CDF U3\U3-CDF-Merged-separate.GROUP.CUT
File Edit Analysis Help
Graphics
Importance
Cumulative Prob.
Density Function
1E-06 1E-05 1E-04 1E-03 1E-02 1E-01 1E+00
Parameter | Estimate | Confidence Range
Point Est | 8.566E-05
Samples | 50000
Mean | 8.544E-05 | [8.4E-05 , 8.7E-05]
5% | 1.348E-05 | [1.3E-05 , 1.4E-05]
Median | 5.095E-05 | [5.0E-05 , 5.1E-05]
95% | 2.578E-04 | [2.5E-04 , 2.6E-04]
StdDev | 1.265E-04
Skewness | 8.373
Smp Size @ 10% | 843
Smp Size @ 2% | 21065
Parameter | Value
Sampling Method | Montecarlo
Sample Size | 50000
Importance Measures | False
Scope | All
Cutset File(s) | C:\Users\ufarradj\AppData\...
Selected Target(s) | U3-CDF-MERGED-SEPARATE
Database | C:\Users\ufarradj\AppData\...
Module File
Plot Increment | AUTO
ACUBE | False

U3 LERF



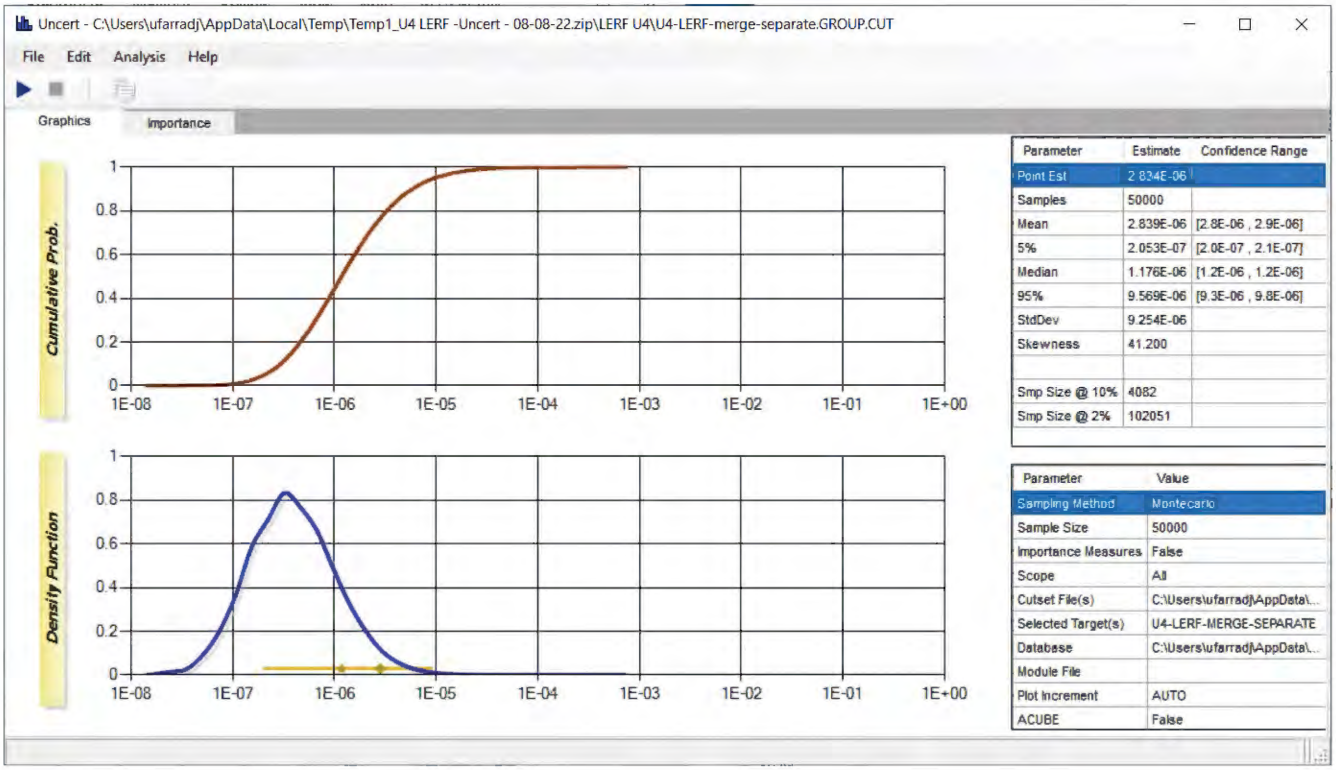
Uncert - C:\Users\ufarradj\AppData\Local\Temp\Temp1_U3 LERF -Uncert - 08-08-22.zip\LERF U3\U3-LERF-merge-separate.GROUP.CUT
File Edit Analysis Help
Graphics
Importance
Cumulative Prob.
Density Function
1E-08 1E-07 1E-06 1E-05 1E-04 1E-03 1E-02 1E-01 1E+00
Parameter | Estimate | Confidence Range
Point Est | 2.971E-06
Samples | 50000
Mean | 3.055E-06 | [2.9E-06 , 3.2E-06]
5% | 1.982E-07 | [1.9E-07 , 2.0E-07]
Median | 1.216E-06 | [1.2E-06 , 1.2E-06]
95% | 1.019E-05 | [9.9E-06 , 1.0E-05]
StdDev | 1.442E-05
Skewness | 78.439
Smp Size @ 10% | 8558
Smp Size @ 2% | 213943
Parameter | Value
Sampling Method | Montecarlo
Sample Size | 50000
Importance Measures | False
Scope | All
Cutset File(s) | C:\Users\ufarradj\AppData\...
Selected Target(s) | U3-LERF-MERGE-SEPARATE
Database | C:\Users\ufarradj\AppData\...
Module File
Plot Increment | AUTO
ACUBE | False

U4 CDF



Uncert - C:\Users\ufarradj\AppData\Local\Temp\Temp1_U4 CDF -Uncert - 08-08-22.zip\CDF U4\U4-CDF-merged-separate.GROUP.CUT
File Edit Analysis Help
Graphics
Importance
Cumulative Prob.
Density Function
1E-06 1E-05 1E-04 1E-03 1E-02 1E-01 1E+00
Parameter | Estimate | Confidence Range
Point Est | 8.971E-05
Samples | 50000
Mean | 9.039E-05 | [8.9E-05 , 9.2E-05]
5% | 1.532E-05 | [1.5E-05 , 1.6E-05]
Median | 5.531E-05 | [5.5E-05 , 5.6E-05]
95% | 2.678E-04 | [2.6E-04 , 2.8E-04]
StdDev | 1.407E-04
Skewness | 14.146
Smp Size @ 10% | 930
Smp Size @ 2% | 23252
Parameter | Value
Sampling Method | Montecarlo
Sample Size | 50000
Importance Measures | False
Scope | All
Cutset File(s) | C:\Users\ufarradj\AppData\...
Selected Target(s) | U4-CDF-MERGED-SEPARATE
Database | C:\Users\ufarradj\AppData\...
Module File
Plot Increment | AUTO
ACUBE | False

U4 LERF


Uncert - C:\Users\ufarradj\AppData\Local\Temp\Temp1_U4 LERF -Uncert - 08-08-22.zip\LERF U4\U4-LERF-merge-separate.GROUP.CUT
File
Edit
Analysis
Help
Graphics
Importance
Cumulative Prob.
Density Function
1E-08
1E-07
1E-06
1E-05
1E-04
1E-03
1E-02
1E-01
1E+00
Parameter | Estimate | Confidence Range
Point Est | 2.834E-06
Samples | 50000
Mean | 2.839E-06 | [2.8E-06 , 2.9E-06]
5% | 2.053E-07 | [2.0E-07 , 2.1E-07]
Median | 1.176E-06 | [1.2E-06 , 1.2E-06]
95% | 9.569E-06 | [9.3E-06 , 9.8E-06]
StdDev | 9.254E-06
Skewness | 41.200
Smp Size @ 10% | 4082
Smp Size @ 2% | 102051
Parameter | Value
Sampling Method | Montecarlo
Sample Size | 50000
Importance Measures | False
Scope | All
Cutset File(s) | C:\Users\ufarradj\AppData\...
Selected Target(s) | U4-LERF-MERGE-SEPARATE
Database | C:\Users\ufarradj\AppData\...
Module File
Plot Increment | AUTO
ACUBE | False

C.2 APPENDIX C REFERENCES

1. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities, EPRI 1011089 - NUREG/CR-6850, August 2005.
2. EPRI/NRC-RES Fire Probabilistic Risk Assessment Methods, Enhancements, Electric Power Research Institute (EPRI), Palo Alto, CA and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD: 2010. EPRI 1019259 and NUREG/CR-6850 Supplement 1.
3. Turkey Point FPRA Summary Report, NUREG/CR-6850 Task 16, Report No. 0493060006.005, Rev. 15, June 2021
4. Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications, ASME/ANS RA-Sa-2009
5. Report PTN-PSA-7.01, Tasks 1 & 6 Plant Partitioning and Ignition Frequencies
6. Electric Power Research Institute (EPRI) and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), "NUREG-2169 Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database United States Fire Event Experience Through 2009," Electric Power Research Institute and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Palo Alto, CA; Washington, D.C., 2015.
7. EPRI/NRC-RES Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, NUREG-2178 EPRI 3002005578, April 2016.
8. Joint Assessment of Cable Damage and Quantification of Effects from Fire (JACQUE-FIRE): Final Report (NUREG/CR-7150, Volume 2: Expert Elicitation Exercise for Nuclear Power Plant Fire-Induced Electrical Circuit Failure), May 2014
9. EPRI/NRC-RES Fire Human reliability Analysis Guidelines – Qualitative Analysis for Main Control Room Abandonment Scenarios, Supplement 1(NUREG-1921), January 2020.
10. Report 0493060006.003. Turkey Point Units 3 and 4 Hot Gas Layer and MultiCompartment Analysis.
11. TURKEY POINT UNITS 3 AND 4 FIRE PROBABILISTIC RISK ASSESSMENT FIRE SCENARIO REPORT NUREG/CR-6850 TASKS 8 AND 11, Report No. H0493060006-2009-080714, Rev 7, June 2021.
12. Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications, NUREG/CR-1278, Dated August 1983.

Appendix D FIRE MODELING WORKBOOK APPROACH

The use of a generic ZOI has a hot gas layer limitation as described in the Generic Fire Modeling Treatments [3] in which the ZOI is increased due to additional heat flux incurred from the hot gas temperature. The PTN Fire PRA implemented the increased ZOI by applying the full room damage target set. The PTN Hot Gas Layer and Multi-Compartment Analysis [7] evaluated the probability of each fire scenario causing a hot gas layer within an enclosed volume. Conversely, risk significant scenarios require refinement in order to evaluate the probability of an ignition source having a reduced ZOI.

The methodology for evaluating the hot gas layer impact and the associated calculation of non-suppression capability was performed by the SDC Tool, a Mathcad based calculation tool.

The Fire Modeling Workbook (FMW) is an Excel based tool used to calculate the probability of target damage and develops NSPs. The FMW incorporates credit for interruptible fires using the NUREG-2230 analysis approach. The FMW was used in the PTN scenario development to analyze and model NSPs for fire scenarios while incorporating the NUREG-2230 approach.

Figure D-1 provides an illustration of the calculation described in the following sections used in the FMW. The two main iterations are maximum simulation heat release rate (noted by variable j), and time (noted by variable i). A time marching simulation is performed for each postulated fire size (known as a bin). Each simulation yields a time to target damage (i.e., critical time). These critical times are utilized by the non-suppression analysis. Finally, a probability of target damage is calculated. The probability associated with each bin is then summed, producing a total probability of target damage.

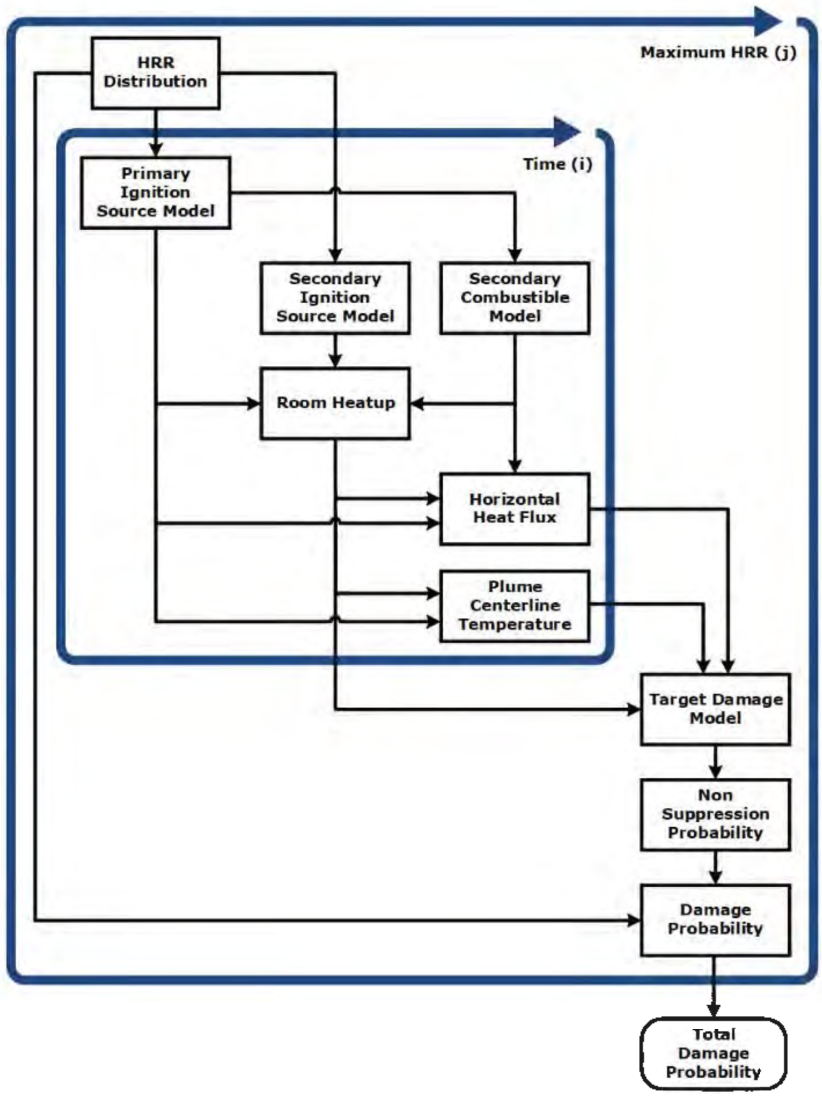


Figure D-1 Conceptual Diagram of the Fire Modeling Workbook Calculation

Refer to the Fire Modeling Workbook Methodology Technical Procedure see Attachment 2 for more details on inputs for FMW and calculations for primary ignition sources, secondary combustibles, enclosure ambient temperature, target damage probability, and scenario validation.

D.1 PTN FMW IMPLEMENTATION

Two FMW databases and corresponding spreadsheets are used, one for the Multi-Compartment Analysis (MCA) scenarios, and one for all other scenarios. The databases manage the input and output from the spreadsheets that perform the scenario NSP calculations. These FMW databases/spreadsheets are provided in Attachment 1.

The FMW incorporates the heat release rate distributions from NUREG-2178 [8], the non-suppression probability values from NUREG-2169 [9], and the interruptible fire approach from NUREG-2230 [10].

Attachment 1 includes the spreadsheets (one for MCA and one for individual ignition sources) that separately calculate the probabilities for the HGL scenarios. The HGL portions of each ignition source and transient fire scenario in a fire zone are summed into one HGL scenario for the fire zone.

A summary of the resulting MCA scenario NSPs is provided in Table 3-1. Table 3-3 provides the fire zone hot gas layer NSPs.

D.2 APPENDIX D REFERENCES

1. Report 0493-06-0006.001, Task 2 Component and Cable Selection Report, Rev 7
2. Report PTN-PSA-7.01, Tasks 1 & 6 Plant Partitioning and Ignition Frequencies, Rev 6
3. Generic Fire Modeling Treatments, Hughes Associates Project Number 1SPH02902.030, Revision 0, January 15, 2008.
4. EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Volume 2; Detailed Methodology. Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD: 2005, EPRI TR-1011989 and NUREG/CR-6850.
5. Hughes Associates Reports 0027-0067-002-005, Rev. 0 for the Unit 3 and Unit 4 Control Room Abandonment Times at the Turkey Point Plant.
6. Fire Events Database and Generic Ignition Frequency Model for U.S Nuclear Power Plants. EPRI, Palo Alto, CA: 2001. 1003111.
7. Report 0493060006.103, Turkey Point Units 1 & 2 Fire Probabilistic Risk Assessment Hot Gas Layer and Multi-Compartment Analysis, Rev 6
8. EPRI/NRC-RES Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, NUREG-2178 EPRI 3002005578, April 2016.
9. Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database, United States Fire Event Experience Through 2009, NUREG-2169 EPRI 3002002936 January 2015
10. NUREG-2230: Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinet Fires in Nuclear Power Plants. EPRI, Palo Alto, CA, and the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Washington, D.C.: 2020. 3002016051/NUREG-2230

Appendix E NSP Calcs Combining NUREG-2180 and NUREG-2330

This appendix describes the methodology and calculations developed to determine the non-suppression probability of selected scenarios at Turkey Point Nuclear Plant when combining the methods presented in NUREG-2230 [1], for interruptible fires, and in NUREG-2180 [2], for incipient detection. The combined methodologies are used for determining non-suppression probabilities for the following fire scenarios:

+ Cable Spreading Room fire scenario with the following characteristics:
    - Ignition Source: Large Enclosure (Group 4a-a), closed, default fuel loading, TP cable
    - Incipient detection system: In-cabinet
    - Redundant Smoke Detection: Ionization detection
    - Time to delayed detection: 15 min
    - Automatic Suppression: Halon System
    - Occupancy: Very Low (Not credited in adjacent zones)
    - Maintenance: Very Low

E.1 SUMMARY NUREG-2230 EVENT TREE

NUREG-2230 describes a detection–suppression event tree that allows for crediting early detection capabilities and personnel suppression capabilities. Recall that the models presented NUREG/CR-6850 [3] or Supplement 1 to NUREG/CR-6850 [4] only credit prompt suppression for fires in the MCR or for fire scenarios associated with hot work activities. The event tree included in NUREG-2230 is a modification of the above models, intended to capture the potential for plant personnel suppression during the early stages of a fire.

The methodology described in NUREG-2230 credits early intervention and suppression by plant personnel by splitting the event tree in NUREG/CR-6850 (for scenarios without incipient detection) into two identical branch groups: one for capturing the non-suppression probability (NSP) for interruptible fires (IF) and one for growing fires (GF). The revised event tree format is presented in Figure E-1.

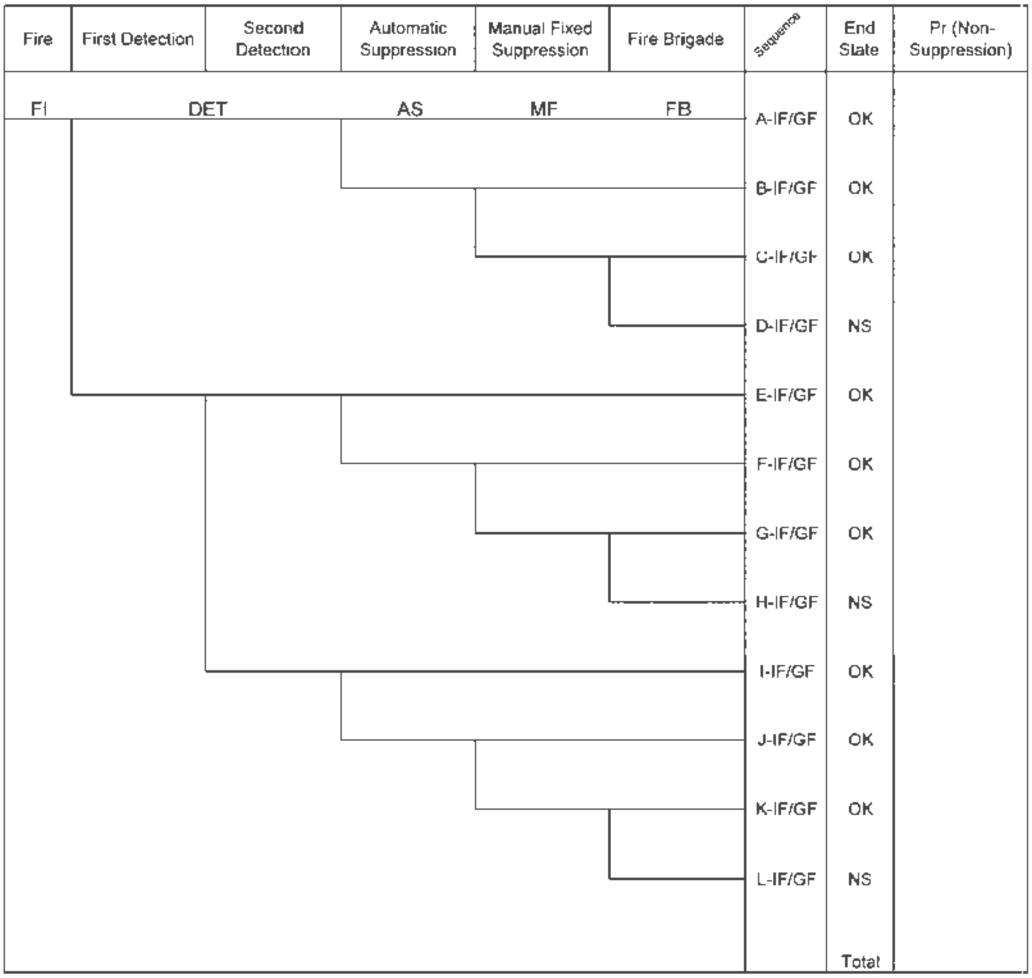


Figure E-1 – Interruptible and Growing Fire Detection and Suppression Event Tree, NUREG-2230.

The revised interruptible and growing sequences (A-L) are conceptually similar to sequences A-N in NUREG/CR-6850. With respect to calculating the NSP for a scenario, required changes are as follows:

+ The probability of detection is no longer split between branches representing the failure of prompt and automatic. Detection is now split between the first detection opportunity (zero time of detection) and the second detection opportunity (modeled time of detection).
+ A unique sequence singling out prompt suppression is no longer included. The development of the interruptible and growing suppression rates makes use of zero detection and short suppression times.
+ The sequence of events associated with delayed detection is retained in this methodology. In NUREG/CR-6850, these sequences were associated with detection by non-automatic means, such as a roving fire watch.
+ It is assumed that a fire will always be detected.
+ The time to detection is assumed to be zero for the following:

  - Detection by a non-fire trouble alarm in the MCR, plant personnel, and automatic smoke detection for an interruptible fire.
  - Detection by a non-fire trouble alarm in the MCR and plant personnel for a growing fire

+ The time to automatic detection for a growing fire may be modeled using the NUREG/CR-6850 growth profile.
+ Special consideration of successful automatic suppression should be taken when included in the interruptible fire path. The interruptible fire introduces the concept of a fire that is not expected to grow to a point that would be capable of activating an automatic suppression system.
+ Similar to a growing fire, the interruptible fire HRR profile, should be used when estimating the activation time of an automatic heat detection or thermally activated automatic suppression system for an interruptible fire.
+ When the NSP for both interruptible and growing fire paths is calculated, the split fraction is applied (and the two probabilities are summed to determine the scenario Pns.

Early detection and suppression by plant personnel is included in the detection-suppression event tree model using the following parameters, which are described in detail in the following sections:

+ Interruptible fire/growing fire split fraction
+ Electrical cabinet HRR timing profiles
+ Automatic (smoke) detection ineffectiveness
+ MCR indication
+ MCR operator response
+ Plant personnel response
+ Plant personnel presence

The approach in NUREG-2230 indicates that for scenarios with incipient detection, the guidance provided in NUREG-2180 should be followed.

E.2 SUMMARY OF NUREG-2180 EVENT TREE

NUREG-2180 developed an event tree to estimate the non-suppression probability for fire scenarios where Very Early Warning Fire Detection (VEWFD) is used. The event tree (Figure 6-4 NUREG-2180, reproduced in Figure E-2 below) estimates the non-suppression probability for in-cabinet smoke detection applications.

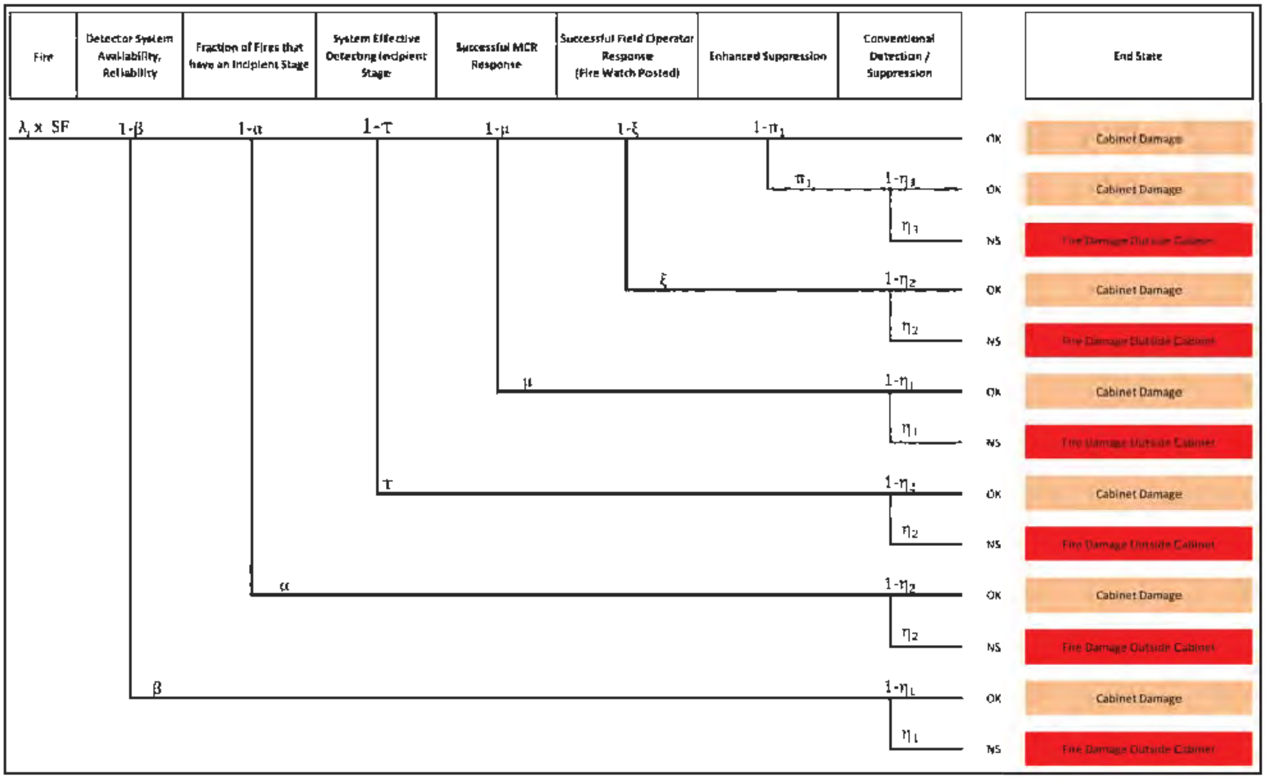


Figure E-2 – Basic Event Tree for In-Cabinet Smoke Detection Non-Suppression Probability Estimation (Figure 6-4 in NUREG-2180).

The event tree headings include estimation of fire phenomena, detector performance, human performance measures, and fire suppression as follows:

- The first event, “Detector System Availability, Reliability” quantifies the systems operational performance. The failure branch (down, β) represents the probability that a detection system will be unable to perform its function because of system outage or hardware failure.
- The next event, “Fractions that have an incipient phase” (α) separates events that exhibit rapidly developing fires from those that exhibit longer incipient stages.
- The next branch “Effectiveness,” evaluates the system’s ability to detect low-energy (pre-flaming) fires for a specific installed application. The success branch (1-τ) represents a detection system’s probability of effectively detecting a low energy fire in its incipient stage. In this case τ represents the smoke detection system’s ineffectiveness in detecting pre-flaming (incipient) phase conditions.
- The human error probability for the MCR operator response is represented by μ. Success of the “MCR Response” event (1-μ) represents that the main control room (MCR) operating crew has acknowledged a smoke detector alert or alarm and has directed first level field response to the alerting/alarming fire location.
- Success in the “First Level Field Response (Technician/Field Operator) Fire Watch Posted” (1-ξ) represents the probability that the field response plant personnel have arrived at the smoke detector alert/alarm location. In this case ξ represents he human error probability for the first level response by the field operator or technician

- Success in the enhanced suppression event ($1-\pi_1$ for in-cabinet detection) represents the probability that any potential fire is suppressed before fire damage to targets of concern.
- The last event "Conventional Detection/Suppression" estimates the probability of successfully suppressing a fire given a failure of one of the earlier events ($1-\eta_1$,$1-\eta_2$ and $1-\eta_3$). To estimate the success of these branches in NUREG-2180 the suppression/detection event tree from NUREG/CR-6850 should be solved for the scenario when redundant detection and/or automatic suppression systems are available in the area as follows:
  - "η_1" represents sequences F – N from the detection suppression event tree in NUREG/CR-6850. That is, given a failure of the VEWFD system or MCR to respond, the redundant detection and/or automatic suppression capability still exists.
  - "η_2" represents sequences F – I from the detection suppression event tree in NUREG/CR-6850. That is, given a failure of the VEWFD system to provide sufficient advance warning, the VEWFD system will still provide prompt detection functions. Time to detection is assumed to be at ignition.
  - "η_3" represents the failure of an independent automatic fire suppression system (including automatic detection system if the automatic suppression system is dependent on the automatic detection system) to suppress the fire prior to fire damage when the enhanced suppression capabilities fail. If an independent automatic suppression system is not present in the fire scenario, then "η_3" is assumed 1.0. For all other cases, the reliability of the independent automatic suppression system (and automatic detection system, if applicable) is modeled consistent with NUREG/CR-6850, including an evaluation of any timing considerations.

E.2.1 Combining NUGEG-2180 with NUREG-2230

As described in Section E.2, the parameters η_1,η_2 and η_3 capture the impact of a conventional detection/suppression system within the NUREG-2180 incipient detection using the detection/suppression event tree presented in Appendix P of NUREG/CR-6850. With the publication of NUREG-2230, the revised framework for the calculation of the non-suppression probability of an electrical cabinet fire may be used to determine the values for parameters η_1,η_2 and η_3.

E.2.1.1 Methodology Differences

There are a number of parameters described in NUREG-2180 and NUREG-2230 that appear to capture similar elements. This section reviews these elements and describes the appropriate use of each when the methods are combined.

1. NUREG-2230 introduced the concepts of "interruptible" and "growing" fires. An "interruptible" fire is one with a relatively slow growth stage that could be: 1) detected, and 2) controlled before growth and propagation outside the ignition source. A "growing" fire refers to faster growing fires that may not be controlled before propagating outside the ignition source. NUREG-2230 recommended a split fraction characterizing the percentage of electrical cabinet fires that may present "interruptible" conditions versus growing conditions. Both the "interruptible" and "growing" fraction of fires may exhibit an incipient phase. That is, the concept of an "interruptible" fire as defined in NUREG-2230 is independent of an ignition source that may present an incipient phase. Therefore, both the fraction of electrical cabinet fires that do not have an incipient phase detectable by an VEWFDS (α) and the fraction of fires have an

incipient phase detectable by an VEWFDS (1-α) should be modeled with the interruptible and growing fire split fractions consistent with the guidance in NUREG-2230.

2. The incipient system in-effectiveness, τ, in the NUREG-2180 methodology and the automatic smoke detection ineffectiveness parameter in the NUREG-2230 methodology are independent. The parameter τ, in the NUREG-2180 is applicable to incipient detection systems. The parameter presented in NUREG-2230 is applicable to automatic smoke detection for flaming fires.

   a) For scenarios where the redundant and independent automatic smoke detection system is located within the electrical cabinet, the ineffectiveness term introduced in NUREG-2230 may be set to zero (0). As described in NUREG-2230, this parameter was introduced to capture the probability of a fire not being capable of producing a detectible signature. This parameter was developed as a function of multiple parameters including fire size and separation of the smoke detector from the fire. It may be assumed that a detector located within the enclosed space of an electrical cabinet while flaming combustion occurs will be sufficient to activate that detector.

3. The successful main control room response parameter, μ, in the NUREG-2180 methodology is independent of the main control room operator response in the NUREG-2230 methodology. In NUREG-2180 this parameter captures the failure of a MCR operator to respond to an incipient fire alarm. In NUREG-2230 this parameter captures the failure of a MCR operator to respond to a non-fire trouble alarm.

4. Credit for personnel detection as described in NUREG-2230 is not negatively impacted in the event of a failure of a VEWFDS. Personnel detection in NUREG-2230 is developed around the likelihood of personnel being present in an area of a fire and is not dependent on the success of an incipient detection system.

E.2.1.2 η_1: Failure of the VEWFDS, Redundant Detection/Suppression Capability

This term captures the event where the incipient detection system has failed or the MCR has failed to recognize the alert. The detection/suppression event trees presented in the NUREG-2230 method can be substituted directly in the NUREG-2180 method to determine the value for η_1 with no modification necessary.

In NUREG-2180 the development of η_1 states that the calculation represents sequences F – N in the NUREG/CR-6850 Appendix P tree. With the introduction of personnel detection in NUREG-2230, the opportunity for what is designated ‘prompt’ detection (sequences A – E in Appendix P of NUREG/CR-6850) is now captured in the first detection step of the NUREG-2230 event tree.

E.2.1.3 η_2: Prompt Alert by VEWFDS, Redundant Detection/Suppression Capability

The term η_2, captures the case where the VEWFDS has not provided advanced warning – detection within the incipient phase – but still provides an alert that allows for crediting ‘prompt’ detection. With the application of NUREG-2230, for this case the probability of first detection should be modeled as 100% successful for both the interruptible and growing fires. Therefore, it is not necessary to apply the automatic smoke detection ineffectiveness parameter, automatic smoke detection unavailability or unreliability, MCR indication, MCR operator response, or the probability that personnel are present. Essentially, the prompt alert by the VEWFDS can be understood to mean the personnel will be in the area of the fire and the probability of personnel present is 100%.

E.2.1.4 η_3: Failure of an Independent Suppression System

There is no change in the application of the parameter that captures the failure of an independent automatic suppression system to suppress a fire prior to damage, η_3, as described in NUREG-2180.

E.2.2 Time to Target Damage

As described in Section E.1.2, the enhanced suppression probability (π) represents the probability that any potential fire is not suppressed before fire damage to targets of concern.

The "π" factor differs between the two event trees in NUREG-2180. The "π_1" factor is applicable for the in-cabinet event tree (see Figure E-2) and represents the probability that, given success of the technician/field operator to respond to the VEWFD "alert," suppression has failed to limit the fire damage to the enclosure of origin. The field operator in the area of the cabinet responsible for the VEWFD system alert fails to promptly suppress the fire quickly enough to prevent damage to PRA targets outside the cabinet. The MCR curve should be used for this case. This is considered to be reasonable representation given that the field operator, a trained responder, will be near the bank of cabinets where the VEWFD system alert was initiated, actively searching for the source location of the alert. The probability of failure to extinguish the fire (π_1), once ignition has occurred, is calculated based on the time available for manual suppression, that in this case is considered the time to target damage (t) as follows:

$$\pi = e^{-\lambda \times t}$$

The time to target damage for each of the percentiles evaluated was provided by the Fire Modeling Workbook, See Appendix D

E.2.3 NUREG-2180 Parameters

This section summarizes the NUREG-2180 parameters used for the scenarios under analysis.

E.2.3.1 Cable Spreading Room - In-Cabinet Detection

Table E-1 lists the NUREG-2180 parameters used for the Cable Spreading Room scenarios with in-cabinet VEWFD. These parameters are the inputs values to the event tree model in Figure 6-4 of NUREG-2180.

Table E-1 – NUREG-2180 Parameters (Cable Spreading Room – In-Cabinet Detection)

Parameter	*Value*	*Justification*
β	3.6E-03	NUREG-2180 (Default Value)
α	2.80E-01	NUREG-2180 (Low Voltage Control Cabinets)
τ	5.3E-01	NUREG-2180 (In-Cabinet – Natural and Forced <100 ACH: ASD LS1)
μ	1E-04	NUREG-2180 (Default Value)

ξ	4.58E-04	NUREG-2180 (In-Cabinet – ASD VEWFD Light Scattering (LS) and VEWFD Light Scattering Sensitive Spot (SS))
λ	0.385	Current MCR Suppression Rate from NUREG- 2178, V2.

E.2.4 NUREG-2230 Parameters

This section summarizes the NUREG-2230 parameters used for the scenarios under analysis.

E.2.4.1 Cable Spreading Room - In-Cabinet Detection

Table E-3 through Table E-5 list the NUREG-2230 parameters used for calculating η_1, η_2, η_3 in the Cable Spreading Room. These parameters are the inputs values to the event tree model in Figure 5-1 of NUREG-2230.

Table E-3 – η_1 Calc Using NUREG-2230 (Cable Spreading Room – In-Cabinet Detection)

Parameter	*Value*	*Justification*
Time to Target damage (min)	Specific value for each percentile	Input data for each percentile from Fire Modeling Workbook
Time to automatic detection (min)	2	Input data
Time to automatic suppression (min)	3	Time to suppression for Halon system assumed 1 minute after automatic detection
Time to delayed detection (min)	15	Input data
Credit automatic detection	TRUE	Input data
Credit automatic suppression	TRUE	Input data
Automatic detection failure probability or unavailability	0.0595	Assuming detection unavailability (0.01) and unreliability (0.05) for smoke detection system.
Automatic (or manual) suppression failure probability or unavailability	0.0595	Assuming suppression unavailability (0.01) and unreliability (0.05) for halon suppression system
Plant personnel activating manual fixed system HEP	-	N/A
Credit MCR Indication	TRUE/FALSE	Assumed no MCR indication
Credit Fixed Manual Supp	FALSE	Input data
Time to target dam, interruptible (min)	Time to Target damage (min) + 4 min	Based on NUREG-2330
Smoke det ineffectiveness	0.2235	Based on NUREG-2330 for Enclosure Class/Function Group: 4a-a (Large, closed, default fuel loading) as input data

Table E-3 – $\eta 1$ Calc Using NUREG-2230 (Cable Spreading Room – In-Cabinet Detection)

Parameter	Value	Justification
Probability of personnel not present in room	0.96709	Based on NUREG-2330 for very low occupancy and maintenance as input data.
Manual suppression probability constant, interruptible	0.149	Based on NUREG-2330 default value
Manual suppression probability constant, growth	0.1	Based on NUREG-2330 default value
Split fraction (% of interruptible fires)	0.723	Based on NUREG-2330 default value
Credit heat detection for interruptible	FALSE	Input data
Time To Auto Detection - IF	Time to automatic detection (min) + 4 min	Based on NUREG-2330
Time To Auto Suppression- IF	Time to automatic detection (min) + 4 min	Based on NUREG-2330

Table E-4 – η2 Calc Using NUREG-2230 (Cable Spreading Room – In-Cabinet Detection)

Parameter	*Value*	*Justification*
Time to Target damage (min)	Specific value for each percentile	From FMW
Time to automatic detection (min)	0	Based on NURE-2230, time to detection for both Interruptible and Growing Fires is considered at t = 0.
Time to automatic suppression (min)	1	Time to suppression for Halon system assumed 1 minute after automatic detection
Time to delayed detection (min)	15	Input data
Credit automatic detection	TRUE	Input data
Credit automatic suppression	TRUE	Input data
Automatic detection failure probability or unavailability	0.0595	Assuming detection unavailability (0.01) and unreliability (0.05) for smoke detection system. As in this case the incipient detector system is considered as personnel always present in the room, this value does not affect the NSP calc.
Automatic (or manual) suppression failure probability or unavailability	0.0595	Assuming suppression unavailability (0.01) and unreliability (0.05) for halon suppression system
Plant personnel activating manual fixed system HEP	-	N/A
Credit MCR Indication	FALSE	Assumed no credit for MCR indication
Credit Fixed Manual Supp	FALSE	Input data
Time to target dam, interruptible (min)	Time to Target damage (min) + 4 min	Based on NUREG-2330
Smoke det ineffectiveness	0	Since the VEWFD is located within the electrical cabinet, as specified in NUREG-2180, given a failure of the VEWFD system to provide sufficient advance warning, the VEWFD system will still provide prompt detection functions

Table E-4 – $\eta 2$ Calc Using NUREG-2230 (Cable Spreading Room – In-Cabinet Detection)

Parameter	*Value*	*Justification*
Probability of personnel not present in room	0	As specified in NUREG2180, given a failure of the VEWFD system to provide sufficient advance warning, the VEWFD system will still provide prompt detection functions. Therefore, in this case, the incipient detector within the electrical cabinet is considered as personnel always present in the room for NUREG-2230.
Manual suppression probability constant, interruptible	0.149	Based on NUREG-2330 default value
Manual suppression probability constant, growth	0.1	Based on NUREG-2330 default value
Split fraction (% of interruptible fires)	0.723	Based on NUREG-2330 default value
Credit heat detection for interruptible	FALSE	Input data
Time To Auto Detection - IF	Time to automatic detection (min) + 4 min	Based on NUREG-2330
Time To Auto Suppression- IF	Time to automatic detection (min) + 4 min	Based on NUREG-2330

Table E-5 – η3 Calc Using NUREG-2230 (Cable Spreading Room – In-Cabinet Detection)

Parameter	*Value*	*Justification*
Suppression Unreliability Halon System	0.05	NUREG/CR-6850
Detection Unreliability Smoke System (Ionization detection)	0.05	NUREG/CR-6850

E.3 CALCULATION PROCESS

The following spreadsheet uses the methodology specified above and incorporate the time to damage from the corresponding fire scenarios in the cable spreading room from the Fire Modeling Workbook. The output from this spreadsheet is an NSP averaged over the probability distribution that is then used in the FRANX model quantification.

See Attachment 3.

APPENDIX E REFERENCES

1. NUREG-2230: Methodology for Modeling Fire Growth and Suppression Response for Electrical Cabinet Fires in Nuclear Power Plants. EPRI, Palo Alto, CA, and the U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Washington, D.C.: 2020. 3002016051/NUREG-2230
2. NUREG-2180: Determining the Effectiveness, Limitations, and Operator Response for Very Early Warning Fire Detection Systems in Nuclear Facilities (DELORES-VEWFIRE). U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD. Final report, December 2016.
3. NUREG/CR-6850: EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities: Volume 2: Detailed Methodology. Electric Power Research Institute (EPRI), Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD: 2005, EPRI TR-1011989 and NUREG/CR-6850.
4. Fire Probabilistic Risk Assessment Methods Enhancements: Supplement 1 to NUREG/CR-6850 and EPRI 1011989. EPRI, Palo Alto, CA, and NRC, Washington, D.C.: December 2009. 1019259.
5. NUREG-2178, V1, and EPRI 3002005578 - Refining And Characterizing Heat Release Rates From Electrical Enclosures During Fire (RACHELLE-FIRE), Volume 1: Peak Heat Release Rates and Effect of Obstructed Plume, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA: 2015.
6. NUREG-2178, V2: Refining and Characterizing Heat Release Rates from Electrical Enclosures During Fire, Volume 2: Fire Modeling Guidance for Electrical Cabinets, Electric Motors, Indoor Dry Transformers, and the Main Control Board, U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Rockville, MD, and Electric Power Research Institute (EPRI), Palo Alto, CA: 2019. NUREG-2178, V2, and EPRI 3002016052.
7. NUREG/CR-7010 Volume 1, “Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) Volume 1: Horizontal Trays,” Final Report, NRC, Washington, DC, July 2012.
8. NUREG/CR-7010 Volume 2, “Cable Heat Release, Ignition, and Spread in Tray Installations During Fire (CHRISTIFIRE) Phase 2: Vertical Shafts and Corridors,” Final Report, NRC, Washington, DC, December 2013.
9. NUREG-2169: Nuclear Power Plant Fire Ignition Frequency and Non-Suppression Probability Estimation Using the Updated Fire Events Database: United States Fire Event Experience Through 2009. EPRI, Palo Alto, CA, and U.S. Nuclear Regulatory Commission, Office of Nuclear Regulatory Research (RES), Washington, D.C.: 2014. 3002002936/NUREG-2169.
10. Casto, G. A., "Close-Out of Fire Probabilistic Risk Assessment Frequently Asked Question 16-0011 on Alternative Methodology to NUREG/CR-6850 for Bulk Cable Tray Ignition Criteria,” ADAMS Accession No. ML18074A020, NRC, Washington, D.C., March 20, 2018.
11. NUREG-1805 - “Fire Dynamics Tools (FDTS) Quantitative Fire Hazard Analysis Methods for the U. S. Nuclear Regulatory Commission Fire Protection Inspection Program,” Iqbal, N. and Salley, M. H., NUREG-1805, Final Report, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D. C., December 2004.
12. NUREG-1805, Supplement 1 “Fire Dynamics Tools (FDTS) Quantitative Fire Hazard Analysis Methods for the U. S. Nuclear Regulatory Commission Fire Protection Inspection Program,” Stroup, D., Taylor, G., Hausman, G., and Salley, M. H., NUREG-1805, Final Report, U.S. Nuclear Regulatory Commission, Office of Nuclear Reactor Regulation, Washington, D. C., July 2013.
13. Heskestad, G. “Fire Plumes, Flame Height, and Air Entrainment,” Section 13, The SFPE Handbook of Fire Protection Engineering, 5th Edition, Society of Fire Protection Engineers (SFPE), Bethesda, ME, 2016.

14. McGrattan, K. and Miles, S., "Modeling Fires Using Computational Fluid Dynamics (CFD)," Section 32, The SFPE Handbook of Fire Protection Engineering, 5th Edition, SFPE, Bethesda, MD, 2016.
15. Inspection Manual Chapter (IMC) 0609, Appendix F, "Fire Protection Significance Determination Process," NRC, Washington, D.C., May 2, 2018.

ATTACHMENTS (No change to attachments in Rev. 1, see Rev. 0 for Attachments)

	File Name	File Size	Date
1	+ Fire Modeling Workbook_PTN_Final.accdb.	19,136 KB	7/25/2022
	+ Fire Modeling Workbook_PTN_FINAL.xlsm	9,480 KB	7/21/2022
	+ Fire Modeling Workbook_MCA_PTN_Final.accdb	20,140 KB	2/28/2022
	+ Fire Modeling Workbook_MCA_PTN_Final.xlsm	9,276 KB	2/25/2022
2	Fire Modeling Workbook Methodology R3.pdf	1,733 KB	11/4/2020
3	Det Supp Event Tree 2180-2230 Not Obs.xlsm	1,862 KB	8/11/2022
4	Steam Generator Overfill Analysis and basic_steaming_calc_for AHFPSGLVL-F_Tsw	1,004 KB	8/12/2022

Turkey Point Nuclear Plant Unit 3 and Unit 4
License Amendment Request 276, Revise Fire Protection Program in Support of Reactor Coolant Pump Seal Replacement Project

ATTACHMENT 4

PASSIVE SHUTDOWN SEAL – EVALUATION OF FAILURE TO ACTUATE FOR FRAMATOME RCP SEALS

(NON-PROPRIETARY VERSION)

(28 pages follow)

Framatome Inc.

Engineering Information Record

Document No.: 51 - 9351505 - 000

Passive Shutdown Seal – Evaluation of Failure To Actuate for Framatome RCP Seals

Framatome Inc.

This document has been redacted and no longer contains Framatome Proprietary Information.

This redacted version is Non-Proprietary.

Proprietary information contained within brackets and proprietary markings have been removed. Other than removal of proprietary information, technical content was not changed as attested to by the following individuals:		
Preparer: Ricky Paugh	**Reviewer:** Philip Opsal	**Approver:** Jonathan Smith
Signature and Date	Signature and Date	Signature and Date

Safety Related? ☐ YES ☒ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, M, A-CRF, A	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
Garrett Snedeker, PRA Supervisor	*GW SNEDEKER 8/12/2022*	P		All
Gordy Salisbury, Risk Informed Engineer	*GC SALISBURY 8/12/2022*	R		All
Ricky Paugh, Engineering Supervisor	*RL PAUGH 8/12/2022*	A		All

Note: P/LP designates Preparer (P), Lead Preparer (LP)
M designates Mentor (M)
R/LR designates Reviewer (R), Lead Reviewer (LR)
A-CRF designates Project Manager Approver of Customer Required Format (A-CRF)
A designates Approver/RTM – Verification of Reviewer Independence

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
N/A			

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial Release

Table of Contents

Page

List of Tables

Page

List of Figures

Page

List of Acronyms

ABWR	Advanced Boiling Water Reactor
AC	Alternating Current
CCCG	Common-Cause Component Group
CCF	Common Cause Failure
CDF	Cumulative Distribution Function
DMA	Dynamic Mechanical Analysis
DSC	Differential Scanning Calorimetric
[	]
ELAP	Extended Loss of AC Power
EDG	Emergency Diesel Generator
FMEA	Failure Modes and Effects Analysis
FTC	Failure to Close
gpm	Gallons per Minute
HCHPF	High Confidence High Probability of Failure
HEP	Human Error Probability
HRA	Human Reliability Analysis
HRAC	HRA Calculator
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
PDF	Probability Density Function
[	]
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSDS	Passive ShutDown Seal
psi	Pounds per Square Inch
PPM	Parts per Million
[	]
PWR	Pressurized Water Reactor
QA	Quality Assurance
QC	Quality Control
RCP	Reactor Coolant Pump
SBO	Station Blackout
THERP	Technique for Human Error-Rate Prediction

1.0 INTRODUCTION

Excessive leakage of reactor coolant pump (RCP) seals following loss of component cooling water or loss of power events can be a significant contributor to risk at nuclear power plants. This is particularly true in the event of an extended station blackout (SBO) or extended loss of alternating current (AC) power (ELAP), when the RCP seals can be exposed to high temperature and high pressure conditions for a significant period of time. To address this issue, Framatome offers a device, the Passive ShutDown Seal (PSDS), which is available as a solution to RCP seal leakage during an extended SBO.

This document presents the results of an evaluation of the failure of the PSDS to actuate during an accident scenario. The evaluation is not based on the application of the PSDS in the seal package for a particular RCP type.

2.0 OBJECTIVE

The objective of this report is to assess the potential failure mechanisms that are associated with the subcomponents of the PSDS, [

]

The objective of the evaluation is to estimate a reasonable probability of the failure to actuate that can be used as part of a probabilistic risk assessment (PRA).

3.0 PSDS DESCRIPTION

The PSDS is a passively-actuated mechanical seal that is designed to provide very low leakage through currently installed RCP seals in the event of ELAP. The PSDS is available pre-assembled into a #1 seal insert that can be installed with little or no modifications to existing RCP seals. The PSDS is installed as an integral portion of the existing #1 seal insert and is located upstream of the No. 1 RCP seal leak-off line. Information on the general design, components, and operation of the PSDS can be found in Reference [1, Section 5].

[

]

[ ]

Figure 3-1: Cross-Section View of PSDS

The PSDS is a passive device, which is not dependent on any support system, e.g., electrical power, cooling water, instrument air, etc. The following is a description of each component of the PSDS and its role in operation of the PSDS.

3.1 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-2 [1, page 18].

Figure 3-2: [ ]

3.2 [ ]

3.3 [ ]

A drawing of a [ ] is shown as Figure 3-3 [1, pages 17, 18].

[ ]

Figure 3-3: [ ]

3.4 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-4 [1, page 19].

Figure 3-4: [ ]

3.5 [ ] / PSDS Assembly

A drawing of the [ ] is shown as Figure 3-5 [1, page 17].

Figure 3-5: [ ]

4.0 ACTUATION OF THE PSDS

[

]

[

]

Figure 4-1: PSDS in Inactive State

[

]

Figure 4-2: PSDS after Actuation

5.0 FAILURE MODES AND EFFECTS ANALYSIS (FMEA)

5.1 Failure to Actuate Failure Modes and Effects Analysis

Table 5-1: FMEA Table – [ ]

[ ]

Table 5-2: FMEA Table – [ ]

[ ]

Table 5-2: FMEA Table – [ ] (Continued)

[ ]

Table 5-3: FMEA Table – [ ]

Table 5-3: FMEA Table – [ ] (Continued)

[ ]

5.2 FMEA Results

After performing the component based FMEA for the PSDS, [ ] The following failure mechanisms are discussed in additional details in the proceeding sections:

[ ]

6.0 EVALUATION OF POTENTIAL FAILURE MECHANISM OF COMPONENTS

6.1 [ ]

[ ]

[ ]

6.2 [ ]

[ ]

6.3 [ ]

[ ]

6.4 [ ]

[ ]

7.0 SUMMARY OF PSDS FAILURE TO ACTUATE

[ ]

7.1 Summary of Basic Events

The following table summarizes all the basic events included in the fault tree that is used to quantify the probability of the PSDS Failure to Actuate.

Table 7-1: Basic Event Summary

[ ]

8.0 REFERENCES

1. Framatome Document No. 38-9351062-000, "Passive Shutdown Seal [ ] for Reactor Coolant Pump [ ]."
2. Framatome Document No. 58-9346852-000, "Passive ShutDown Seal (PSDS) for Reactor Coolant Pump - [ ] - Test Report"
3. AREVA Document No. 38-9227792-000, "Dispositif d'Etancheite Passif pour systeme de joints d'arbre des [ ], Note de Synthese de Qualification (lot N1)."

APPENDIX A: FAILURE OF PSDS TO ACTUATE FAULT TREE

The following is the fault tree developed to calculate the probability of the failure of the PSDS to actuate.

APPENDIX B: HRA CALCULATOR DETAILS

PSDS-ASSEMBLY,

Turkey Point Nuclear Plant Unit 3 and Unit 4
License Amendment Request 276, Revise Fire Protection Program in Support of Reactor Coolant Pump Seal Replacement Project

ATTACHMENT 6

PASSIVE SHUTDOWN SEAL – EVALUATION OF FAILURE TO REMAIN SEALED

(NON-PROPRIETARY VERSION)

(36 pages follow)

Framatome Inc.

Engineering Information Record

Document No.: 51 - 9348566 - 001

Passive Shutdown Seal – Evaluation of Failure To Remain Sealed

Framatome Inc.

This document has been redacted and no longer contains Framatome Proprietary Information.

This redacted version is Non-Proprietary.

Proprietary information contained within brackets and proprietary markings have been removed. Other than removal of proprietary information, technical content was not changed as attested to by the following individuals:		
Preparer: Ricky Paugh	**Reviewer:** Philip Opsal	**Approver:** Jonathan Smith
Signature and Date	Signature and Date	Signature and Date

Safety Related? ☐ YES ☒ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, M, A-CRF, A	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
Garrett Snedeker, PRA Supervisor	GW SNEDEKER 8/15/2022	P		All
Gordy Salisbury, Risk Informed Engineer	GC SALISBURY 8/15/2022	R		All
Ricky Paugh, Engineering Supervisor	RL PAUGH 8/15/2022	A		All

Note: P/LP designates Preparer (P), Lead Preparer (LP)
M designates Mentor (M)
R/LR designates Reviewer (R), Lead Reviewer (LR)
A-CRF designates Project Manager Approver of Customer Required Format (A-CRF)
A designates Approver/RTM – Verification of Reviewer Independence

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
N/A			

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial Release
001	All	Minor Revision throughout

Table of Contents

Page

List of Tables

Page

List of Figures

Page

List of Acronyms

AC	Alternating Current
CCCG	Common-Cause Component Group
CCF	Common Cause Failure
CDF	Cumulative Distribution Function
DMA	Dynamic Mechanical Analysis
DSC	Differential Scanning Calorimetric
[	]
ELAP	Extended Loss of AC Power
EDG	Emergency Diesel Generator
FMEA	Failure Modes and Effects Analysis
FTC	Failure to Close
gpm	Gallons per Minute
HEP	Human Error Probability
HRA	Human Reliability Analysis
HRAC	HRA Calculator
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
PDF	Probability Density Function
[	]
PORV	Power Operated Relief Valve
PRA	Probabilistic Risk Assessment
PSDS	Passive ShutDown Seal
psi	Pounds per Square Inch
PPM	Parts per Million
[	]
PWR	Pressurized Water Reactor
QA	Quality Assurance
QC	Quality Control
RCP	Reactor Coolant Pump
SBO	Station Blackout
THERP	Technique for Human Error-Rate Prediction

1.0 INTRODUCTION

Excessive leakage of reactor coolant pump (RCP) seals following loss of component cooling water or loss of power events can be a significant contributor to risk at nuclear power plants. This is particularly true in the event of an extended station blackout (SBO) or extended loss of alternating current (AC) power (ELAP), when the RCP seals can be exposed to high temperature and high pressure conditions for a significant period of time. To address this issue, Framatome offers a device, the Passive ShutDown Seal (PSDS), which is available as a solution to RCP seal leakage during an extended SBO.

This document presents the results of an evaluation of the failure of the PSDS to remain sealed after successful actuation. The evaluation is not based on the application of the PSDS in the seal package for a particular RCP type.

2.0 OBJECTIVE

The objective of this report is to assess the potential failure mechanisms that are associated [

]

The objective of the evaluation is to estimate a reasonable probability of the failure to actuate that can be used as part of a probabilistic risk assessment (PRA).

3.0 PSDS DESCRIPTION

The PSDS is a passively-actuated mechanical seal that is designed to provide very low leakage through currently installed RCP seals in the event of ELAP. The PSDS is available pre-assembled into a #1 seal insert that can be installed with little or no modifications to existing RCP seals. The PSDS is installed as an integral portion of the existing #1 seal insert and is located upstream of the No. 1 RCP seal leak-off line. Information on the general design, components, and operation of the PSDS can be found in Reference [1, Section 5].

[

]

[ ]

Figure 3-1: Cross-Section View of PSDS

The PSDS is a passive device, which is not dependent on any support system, e.g., electrical power, cooling water, instrument air, etc. The following is a description of each component of the PSDS and its role in operation of the PSDS.

3.1 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-2 [1, page 18].

[ ]

Figure 3-2: [ ]

3.2 [ ]

[ ]

3.3 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-3 [1, pages 17, 18].

[ ]

Figure 3-3: [ ]

3.4 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-4 [1, page 19].

Figure 3-4: [ ]

3.5 [ ] / PSDS Assembly

A drawing of the [ ] is shown as Figure 3-5 [1, page 17].

Figure 3-5: [ ]

4.0 ACTUATION OF THE PSDS

[

] A schematic showing a PSDS in the inactive state is shown as Figure 4-1 [3, page 16].

[

]

Figure 4-1: PSDS in Inactive State

[

] A schematic showing a PSDS in the actuated state is shown as Figure 4-2 [3, page 16].

Figure 4-2: PSDS after Actuation

5.0 FAILURE MODES AND EFFECTS ANALYSIS (FMEA)

5.1 Failure to Actuate Failure Modes and Effects Analysis

Table 5-1: FMEA Table – [ ]

[ ]

5.2 FMEA Results

After performing the component based FMEA for the Sealing Split Ring, [ ] The following failure mechanisms are discussed in additional details in the proceeding sections:

[ ]

6.0 EVALUATION OF POTENTIAL FAILURE MECHANISM OF COMPONENTS

The following section describes the potential failure mechanisms of the [ ]. These failures are quantified using the fault tree included in Figure 6-1. Table 7-1 summarizes the basic event probabilities and basis for the values.

[ ]

Figure 6-1: Failure of PSDS To Remain Sealed Fault Tree

6.1 Temperature Related Failure

6.1.1 [ ]

6.1.2 [ ]

6.1.3 [ ]

6.2 [ ]

6.3 [ ]

6.4 [ ]

6.5 [ ]

7.0 SUMMARY OF PSDS FAILURE TO ACTUATE

7.1 Summary of Basic Events

The following table summarizes all the basic events included in the fault tree that is used to quantify the probability of the PSDS Failure to Remain Sealed. See Appendix B for the fault tree used to quantify the failure rate of the PSDS to remain sealed.

Table 7-1: Basic Event Summary

[ ]

8.0 REFERENCES

1. Framatome Document No. 38-9351062-000, "Passive Shutdown Seal [ ] for Reactor Coolant Pump [ ]."

2. Framatome Document No. 58-9346852-000, "Passive ShutDown Seal (PSDS) for Reactor Coolant Pump - [ ] - Test Report"

3. AREVA Document No. 38-9227792-000, "Dispositif d'Etancheite Passif pour systeme de joints d'arbre des [ ], Note de Synthese de Qualification (lot N1)."

APPENDIX A: [ ] TECHNICAL DATA SHEET

APPENDIX B: FAILURE OF PSDS TO REMAIN SEALED FAULT TREE

The following is the fault tree developed to calculate the probability of the failure of the PSDS to remain sealed after successful actuation.

APPENDIX C: HRA CALCULATOR DETAILS

PSDS-ASSEMBLY,

[ ]

[

]

[ ]

Turkey Point Nuclear Plant Unit 3 and Unit 4
License Amendment Request 276, Revise Fire Protection Program in Support of Reactor Coolant Pump Seal Replacement Project

ATTACHMENT 8

PASSIVE SHUTDOWN SEAL – EVALUATION OF SPURIOUS ACTUATION

(NON-PROPRIETARY VERSION)

(38 pages follow)

framatome

Framatome Inc.

Engineering Information Record

Document No.: 51 - 9227814 - 004

Passive Shutdown Seal – PRA Evaluation of Spurious Actuation

Framatome Inc.
This document has been redacted and no longer contains Framatome Proprietary Information.
This redacted version is Non-Proprietary.

Proprietary information contained within brackets and proprietary markings have been removed. Other than removal of proprietary information, technical content was not changed as attested to by the following individuals:		
Preparer: Ricky Paugh	**Reviewer:** Philip Opsal	**Approver:** Jonathan Smith
Signature and Date	Signature and Date	Signature and Date

Safety Related? ☐ YES ☒ NO

Does this document establish design or technical requirements? ☐ YES ☒ NO

Does this document contain assumptions requiring verification? ☐ YES ☒ NO

Does this document contain Customer Required Format? ☐ YES ☒ NO

Signature Block

Name and Title/Discipline	Signature	P/LP, R/LR, M, A-CRF, A	Date	Pages/Sections Prepared/Reviewed/ Approved or Comments
Ahmed Abdelnabi Engineer	*A ABDELNABI 8/15/2022*	LP		All
Garrett Snedeker PRA Supervisor	*GW SNEDEKER 8/15/2022*	LR		All
Ricky Paugh Engineering Supervisor	*RL PAUGH 8/15/2022*	A		All

Note: P/LP designates Preparer (P), Lead Preparer (LP)
M designates Mentor (M)
R/LR designates Reviewer (R), Lead Reviewer (LR)
A-CRF designates Project Manager Approver of Customer Required Format (A-CRF)
A designates Approver/RTM – Verification of Reviewer Independence

Project Manager Approval of Customer References (N/A if not applicable)

Name (printed or typed)	Title (printed or typed)	Signature	Date
N/A			

Record of Revision

Revision No.	Pages/Sections/ Paragraphs Changed	Brief Description / Change Authorization
000	All	Initial Release
001	Section 3.0	Edited text to identify location of PSDS and remove sentences associated with one of the references.
	FMEA Table	Changed table number.
002	Throughout document	Changed Areva to Framatome.
	Throughout document	Individual reference locations were updated in accordance with new updated references.
	Table of Contents	Section page numbers are updated.
	Section 1.0	Scope of document is 100A RCP.
	Section 6.1.2	Table numbers and figure numbers updated.
	Section 6.2	Updated service life of PSDS from [ ] to twelve years [ ]
	Section 6.5	Estimation of spurious actuation frequency changed based on 12 year service life.
	Section 7.0	[ ]
	Section 8.0	Updated References.
003	Throughout document	Revised the document so that it encompasses Low Temperature PSDS [ ]
	Throughout Document	Updated References
	Appendix A	Replaced [ ] Data Sheet with a new one.
	Appendix B	Added Product Data Sheet for [ ]
	Appendix C	Added Framatome QA/QC requirements for RCP Seal Parts

Table of Contents

Page

Table of Contents
(continued)

List of Tables

List of Figures

Page

List of Acronyms

AC	Alternating Current
CDF	Cumulative Distribution Function
DMA	Dynamic Mechanical Analysis
DSC	Differential Scanning Calorimetric
[	]
ELAP	Extended Loss of AC Power
EDG	Emergency Diesel Generator
FMEA	Failure Modes and Effects Analysis
FTC	Failure to Close
gpm	Gallons per Minute
LOCA	Loss of Coolant Accident
LOOP	Loss of Off-Site Power
PDF	Probability Density Function
[	]
PRA	Probabilistic Risk Assessment
PSDS	Passive ShutDown Seal
psi	Pounds per Square Inch
PPM	Parts per Million
[	]
PWR	Pressurized Water Reactor
QA	Quality Assurance
QC	Quality Control
RCP	Reactor Coolant Pump
SBO	Station Blackout

1.0 INTRODUCTION

Excessive leakage of reactor coolant pump (RCP) seals following loss of component cooling water or loss of power events can be a significant contributor to risk at nuclear power plants. This is particularly true in the event of an extended station blackout (SBO) or extended loss of alternative current (AC) power (ELAP), when the RCP seals can be exposed to high temperature and high-pressure conditions for a significant period of time. To address this issue, Framatome offers a device, the Passive ShutDown Seal, (PSDS), which is available as a solution to RCP seal leakage during an extended SBO or ELAP condition.

This document presents the results of an evaluation of the potential for a PSDS to spuriously actuate. The evaluation is for the application of the PSDS in the seal package for [

]

2.0 OBJECTIVE

The objective of this report is to assess the potential failure mechanisms that are associated with [

]

[

]

The objective of the evaluation is to estimate a reasonable probability (frequency) of spurious actuation that can be used as part of a probabilistic risk assessment (PRA).

3.0 PSDS DESCRIPTION

The PSDS is a passively-actuated mechanical seal that is designed to provide very low leakage through currently installed RCP seals in the event of ELAP. The PSDS is available pre-assembled into a #1 seal insert that can be installed with little or no modifications to existing RCP seals. The PSDS is installed as an integral portion of the existing #1 seal insert and is located upstream of the No. 1 RCP seal leak-off line. Information on the general design, components, and operation of the PSDS can be found in Reference 2, Section 5.

[

]

Figure 3-1: PSDS Assembly View

The PSDS is a passive device, which is not dependent on any support system, e.g., electrical power, cooling water, instrument air, etc. The following is a description of each component of the PSDS and its role in operation of the PSDS.

The PSDS comes in two models, a high temperature model, and a low temperature model. The high temperature model actuates at a higher temperature than the low temperature model. The difference in actuation temperature between the two models is due to different fuse spacer material, explained in further detail in Section 3.1.

3.1 [ ]

A drawing of a [ ] is shown as Figure 3-1, [2, page 18].

Figure 3-2: [ ]

3.2 [ ]

3.3 [ ]

A drawing of a PSDS [ ] is shown as Figure 3-2 [2, pages 17, 18].

Figure 3-3: [ ]

3.4 [ ]

[ ]

A drawing of a [ ] is shown as Figure 3-3 [2, page 19].

[ ]

Figure 3-4: [ ]

3.5 [ ] / PSDS Assembly

[ ]

A drawing of the [ ] is shown as Figure 3-4 [2, page 17].

Figure 3-5: [ ]

4.0 ACTUATION OF THE PSDS

[

] A schematic showing a PSDS in the inactive state is shown as Figure 4-1 [3, page 16].

Figure 4-1: PSDS in Inactive State

[

] A schematic showing a PSDS in the actuated state is shown as Figure 4-2 [3, page 16].

[

]

Figure 4-2: PSDS after Actuation

5.0 FAILURE MODES AND EFFECTS ANALYSIS (FMEA)

5.1 Spurious Actuation Failure Modes and Effects Analysis

[

]

A tabular-format FMEA for the fuser spacer was developed as shown in Table 5-1. The FMEA considered the function, possible failure modes, failure mechanisms, and possible preventative measures for spurious actuation of the PSDS due to [

]

Table 5-1: FMEA Table – [ ]

[ ]

5.2 FMEA Results

After performing the component-based FMEA for [ ]

[ ]

6.0 EVALUATION OF MECHANICAL STRESS AND STRENGTH OF [ ]

6.1 [ ]

[ ]

6.1.1 [ ]

[ ]

[ ]

Figure 6-1: [ ]

6.1.2 [ ]

[ ]

Table 6-1: [ ]

[ ]

Table 6-2: [ ]

[ ]

[ ]

Figure 6-2: [ ]

[ ]

[ ]

Figure 6-3: [ ]

6.2 [ ]

[ ]

6.3 [ ]

[ ]

[ ]

Figure 6-4: [ ]

6.4 [ ]

[ ]

Figure 6-5: [ ]

[

]

6.5 Result

[

]

7.0 EVALUATION OF [
]

[

]

Figure 7-1: [ ]

7.1 [ ]

7.2 [ ]

[ ]

7.2.1 [ ]

[ ]

Table 7-1: [ ]

[ ]

7.2.2 [ ]

[ ]

8.0 REFERENCES

1. AREVA Document No. 58-9217959-000, "Passive ShutDown Seal (PSDS) for Reactor Coolant Pump, Qualification of the [ ] Test Report – Impact of the PSDS Inadvertent Actuation on the [ ]"
2. Framatome Document No. 38-9217970-002, "Passive ShutDown Seal (PSDS) for Reactor Coolant Pump, Qualification of the [ ] – Summary Report (US Unit)."
3. AREVA Document No. 38-9227792-000, "Dispositif d'Etancheite Passif pour systeme de joints d'arbre des [ ], Note de Synthese de Qualification (lot N1)."
4. Ichikawa, M., "A Meaning of the Overlapped Area Under Probability Density Curves of Stress and Strength," *Reliability Engineering and System Safety* **41** (1993) 203-204.
5. AREVA Document No. 58-9217956-001, "Passive ShutDown Seal (PSDS) for Reactor Coolant Pump, Qualification of the [ ] Results of the Aging and Corrosion Resistance Tests on [ ]."
6. [ ]
7. PlasticsEurope, 2005, "Guide for the Safe Handling of Fluoropolymer Resins," 3rd Edition, June 2005.
8. AREVA Document No. 38-9227846-000, "Vitesse de rotation de la ligne d'arbre en phase d'activation du dispositive (lot E1), Note d'Analyse."
9. Framatome Document No. 38-9351062-000, "Turkey Point NPP units 3 & 4, Passive Shutdown Seal model [ ], Qualification Report."
10. [ ]
11. AREVA Document No. 58-9350783-000, "Passive Shutdown Seal (PSDS) for Reactor Coolant Pump, Qualification of the [ ] Report of the aging and corrosion resistance tests of the [ ]."

APPENDIX A: [ ] TECHNICAL DATA SHEET

APPENDIX B: [ ] TECHNICAL DATA SHEET

[ ]

APPENDIX C: FRAMATOME INC. RCP SEALS QA/QC REQUIREMENTS

APPENDIX D: DOCUMENTATION OF SPREADSHEET USED TO DETERMINE PDF

[ ]