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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 4a: Reactor, At-Power, Level 1 PRA for Internal Fires

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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Volume 4a: Reactor, At-Power, Level 1 PRA for Internal Fires

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ABSTRACT

The U.S. Nuclear Regulatory Commission performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The scope of the L3PRA project encompasses all major radiological sources on the site (i.e., reactors, spent fuel pools, and dry cask storage), all internal and external hazards, and all modes of plant operation. A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources. This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-power, Level 1 PRA model for internal fires for a single unit. The analyses documented herein are based information for the reference plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained.[1]

CAUTION: While the L3PRA project is intended to be a state-of-practice study, due to limitations in time, resources, and plant information, some technical aspects of the study were subjected to simplifications or were not fully addressed. As such, inclusion of approaches in the L3PRA project documentation should ***not*** be viewed as an endorsement of these approaches for regulatory purposes.

[1] An overview report, which covers all three PRA levels, has been created for each major element of the L3PRA project scope (e.g., for the combined internal event and internal flood PRAs for a single reactor unit operating at full power). These overview reports include a reevaluation of plant risk based on a set of updated plant equipment and PRA model assumptions (e.g., incorporation of the current reactor coolant pump shutdown seal design at the reference plant and the potential impact of the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies [FLEX], both of which reduce the risk to the public).

FOREWORD

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities," dated July 7, 2011 (ML11090A039).

Licensee information used in performing the Level 3 PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models used in its regulatory decision-making. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

Each set of L3PRA project reports covering the Level 1, 2, and 3 PRAs for a specific site radiological source, plant operating state, and hazard group is accompanied by an overview report. The overview reports summarize the results and insights from all three PRA levels.

To provide results and insights better aligned with the current design and operation of the reference plant, the overview reports also provide the results of a parametric sensitivity analysis based on a set of new plant equipment and PRA model assumptions for all three PRA levels. The sensitivity analysis reflects the current reactor coolant pump shutdown seal design at the reference plant, as well as the potential impact of FLEX strategies,[1] both of which reduce the risk to the public.

A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources (i.e., spent fuel in pools and dry storage casks). These insights may be used to further enhance the regulatory framework and decision-making and to help focus limited agency resources on issues most directly related to the agency's mission to protect public health and safety. More specifically, potential future uses of the L3PRA project can be categorized as follows (a more detailed list is provided in SECY-12-0123, "Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC's Regulatory Framework," dated September 13, 2012 [ML12202B170]):

- enhancing the technical basis for the use of risk information (e.g., obtaining updated and enhanced understanding of plant risk as compared to the Commission's safety goals)

- improving the PRA state-of-practice (e.g., demonstrating new methods for site risk assessments, which may be particularly advantageous in addressing the risk from

[1] FLEX refers to the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

advanced reactor designs, a multi-unit accident, or an accident involving spent fuel; and using PRA information to inform emergency planning)

- identifying safety and regulatory improvements (e.g., identifying potential safety improvements that may lead to either regulatory improvements or voluntary implementation by licensees)

- supporting knowledge management (e.g., developing or enhancing in-house PRA technical capabilities)

In addition, the overall L3PRA project model can be exercised to provide insights regarding other issues not explicitly included in the current project scope (e.g., security-related events or the use of accident tolerant fuel). Furthermore, some future advanced light water reactor (ALWR) and advanced non-light water reactor (NLWR) applicants may rely heavily on the results of analyses similar to those used in the L3PRA project to establish their licensing basis and design basis by using the Licensing Modernization Project (LMP) (NEI 18-04, Rev. 1) which was endorsed via Regulatory Guide 1.233 in June 2020. Licensees who use the LMP framework are required to perform Level 3 PRA analyses. Therefore, another potential use of the methodology and insights generated from this study is to inform regulatory, policy, and technical issues pertaining to ALWRs and NLWRs.

The results and perspectives from this report, as well as all other reports prepared in support of the L3PRA project, will be incorporated into a summary report to be published after all technical work for the L3PRA project has been completed.

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EXECUTIVE SUMMARY

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, “Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities,” dated July 7, 2011 (ML11090A039).

As described in SECY-11-0089, the objectives of the L3PRA project are the following:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data, that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (ML040140729), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown risk, multi-unit risk, other radiological sources).

- Extract new insights to enhance regulatory decision making and to help focus limited NRC resources on issues most directly related to the agency’s mission to protect public health and safety.

- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.

- Demonstrate technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs.

This report documents the single-unit, reactor at-power, Level 1 internal fire PRA (FPRA) that supports the L3PRA project. Licensee information used in performing the L3PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. (For example, the L3PRA does not reflect the current reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.[1]) In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models. Finally, it should be noted that the FPRA for the L3PRA project does not account for recent and ongoing work in areas where realism in fire PRA can be improved (e.g., more realistic models of cable damage and functionality, improved heat release rate distributions, better fire modeling techniques, and fire model validation). **As such, this report will not be the sole basis for any regulatory decisions specific to the reference plant.**

As a part of the L3PRA study, the NRC already constructed a Level 1, at-power, PRA model for internal events using the Systems Analysis Programs for Hands-on Integrated Reliability

[1] FLEX refers to the U.S. nuclear power industry’s proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

Evaluations (SAPHIRE) software. This Level 1 internal event PRA model is used as the starting point for the internal fire PRA model. The Level 1 internal event PRA model contains all the potential initiating events, plant response via event tree development, and individual mitigating system response via fault trees. The Level 1 internal event PRA model also contains the success criteria for each of the mitigating systems and the overall plant mission time.

Some areas of the Level 1 internal event PRA model were expanded to account for addition failures that are specific to internal fire events. This expansion involved adding a new event tree for each fire scenario. Additional fault trees were added to account for spurious operations of systems and components that were identified and modeled in the reference plant FPRA, along with failure data to account for spurious operation. The fire-related equipment failures are identified in target sets that are associated with each fire scenario.

The reference plant FPRA has been peer reviewed and the provided documents address the fire PRA tasks listed in NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities.” The NRC commissioned an independent review of the reference plant FPRA. The review was performed to determine the adequacy of the models based on the identified tasks in NUREG/CR-6850. This review was performed by Sandia National Laboratories (SNL). Several items of interest are identified in this review. These items, their impact, and their potential treatment are discussed in the relevant sections of this report.

This report is organized in accordance with the tasks included in NUREG/CR-6850. In many cases, the tasks were not performed for the L3PRA project FPRA; rather, based on the results of the independent review, the reference plant information and analyses were used directly in the L3PRA project FPRA. In other cases, the L3PRA project team performed an adaptation of the work performed by the reference plant or performed the task independently.

The reference plant FPRA includes 3,306 fire sequences, which, for practicality, were mapped into 210 fire scenarios in the L3PRA project FPRA. The NRC staff and its contractors performed several comparisons to confirm that the mapping approach did not unduly inflate the fire core damage frequency (CDF).

The CDF for the L3PRA project FPRA was calculated to be 6.1×10^{-5} per reactor critical year (rcy). There are 31 fire scenarios that contribute 1 percent or more to the overall CDF and have a cumulative CDF of 3.4×10^{-5}/rcy (56 percent of total fire CDF). The top 10 fire scenarios contribute 29 percent of the total fire CDF (1.8×10^{-5}/rcy).

Development and quantification of the L3PRA project FPRA led to several key insights related to the following topics:

- Analysis of fire compartments
- Fire-induced and conditional loss of offsite power (LOOP)
- Spurious equipment actuations and valve transfers
- Consequential small loss-of-coolant accidents (LOCAs)
- Multi-compartment fire analysis
- Risk-significant failure events

Analysis of fire compartments

The reference plant FPRA identified 443 fire compartments. When ranked by CDF, the top 50 fire compartments contribute 90.1 percent of the total fire CDF in the L3PRA project FPRA. As is typical in most fire PRAs, the fire compartments that contribute the most to CDF include the main control room, the switchgear rooms, and the cable spreading rooms. In terms of individual buildings/structures located at the site, the control building contributes the most to total fire CDF at 72 percent. The contributions from the other buildings to total fire CDF include: the auxiliary building (7 percent), the Unit 1 turbine building (4 percent), the Unit 1 containment building (5 percent), the Unit 2 buildings grouped together (5 percent),[2] and the remaining buildings and yard (6 percent).

Fire-induced and conditional LOOP

The results of the L3PRA project FPRA indicate that fire-induced and conditional LOOP events are significant contributors to fire CDF. However, due to the complexity of the modeling, the exact contribution of fire-induced and conditional LOOP to CDF is difficult to ascertain. Using various techniques, fire-induced and conditional LOOP CDF was estimated to be approximately 2×10^{-5}/rcy (~32 percent of total fire CDF).

In the L3PRA at-power Level 1 PRA model for internal events, alignment of the alternate source of offsite power is credited for switchyard-centered and plant-centered LOOP events, as well as transient-induced LOOP events, given failure of onsite power (i.e., emergency diesel generators). This same crediting of the operator action to align the alternate source of offsite power is used in the L3PRA project FPRA, since the Level 1 at-power internal events model and assumptions are used as the starting point. Total fire CDF would increase by approximately 49 percent if credit were not taken for aligning the alternate source of offsite power.

Spurious equipment actuations and valve transfers

Spurious equipment actuations (in particular, multiple spurious operations) that can result from a fire are important contributors to fire CDF. Spurious equipment actuations are estimated to contribute 1.6×10^{-5}/rcy to overall fire CDF (26 percent).

The component whose spurious actuation contributes the most to fire-induced spurious actuation CDF is the steam supply valve to the turbine-driven auxiliary feedwater (AFW) pump. The spurious opening of this valve will cause the pump to start and can lead to overfilling the steam generators and an induced steam line break. This valve contributes 36 percent to the spurious actuation CDF. The next largest contributors to spurious actuation CDF are the pressurizer PORVs. Spurious opening of the PORVs will cause a consequential LOCA (either small or medium, depending on whether one or both valves spurious open), thereby requiring reactor coolant makeup. The dominant accident sequences involving spurious operation of the PORVs also involve fire-induced damage to the makeup capability, which leads to the importance of these components. Collectively, spurious actuation of the PORVs contributes 23 percent to spurious actuation CDF.

[2] As discussed in the reference plant FPRA documentation, Unit 2 fires that can impact Unit 1 include fires originating in the shared control room or fires originating in Unit 2 fire compartments that contain offsite power cables that can damage one or both transformers from Unit 1.

The results in the previous paragraph are based on looking at the spurious operation of components as a group, whether the identified component failed individually as a single spuriously operated component within a cut set or whether it was part of multiple spuriously operated components within a cut set. The overall group of cut sets involving one or more spurious operations was parsed to obtain just those cut sets that contained multiple spurious basic events. The subset of cut sets involving multiple spurious operations has a combined CDF of 1.5×10^{-6}/rcy, which contributes approximately 10 percent to spurious actuation CDF. An example of multiple spurious operations that lead directly to core damage is the spurious opening of a PORV (resulting in a small LOCA) and spurious closing of the RWST suction valve for the safety injection and charging pumps.

Consequential small LOCAs

Consequential small LOCAs are significant contributors to fire CDF. A consequential SLOCA can result from a stuck open or spuriously opened power-operated relief valve (PORV) or a reactor coolant pump (RCP) seal LOCA. Collectively, consequential small LOCAs contribute 23 percent to the total fire CDF (1.4×10^{-5}/rcy). This contribution is dominated by RCP seal LOCAs. The loss of RCP seal cooling is most likely to occur due to loss of either the auxiliary component cooling water system or nuclear service cooling water system. These two systems are also needed to support long-term reactor coolant makeup capability. Therefore, if either of these systems fails, the occurrence of an RCP seal LOCA, via either seal failure or operator failure to trip the RCPs, will lead directly to core damage. In addition, the probabilities of RCP seal failure and operator failure to trip the RCPs are relatively large (0.21 and 0.33, respectively).

Multi-compartment fire analysis results

The reference plant performed an extensive multi-compartment fire analysis (MCA). Ten MCA sequences survived the screening process and were evaluated in the reference plant FPRA. However, only two MCA scenarios were modeled in the L3PRA project FPRA because the other eight fell below the reference plant FPRA truncation limit. The CDF from the two remaining MCA scenarios is 1.3×10^{-7}/rcy, which is a small fraction of the total L3-FPRA CDF. This result supports the statement in the reference plant FPRA documentation that “the [reference plant units] are very well compartmentalized with most boundaries containing fire rated barriers. Therefore, multi compartment fires have a negligible impact on total plant risk.”

Risk-significant failure events

Human failure events (HFEs) are major contributors to core damage in the L3PRA project FPRA. This is due to the limited mitigating systems available in many fire scenarios. The HFE with the highest percentage contribution to CDF (24 percent) is operator failure to initiate bleed and feed cooling in the absence of secondary-side heat removal. This action either occurs as an independent operator failure or in cut sets with the HFE for failure to control AFW, for which there is a moderate dependency.

The HFE with the second highest percentage contribution to CDF (11 percent) is operator failure to trip the RCPs given loss of seal cooling. Per the WOG-2000 RCP seal LOCA model, failure to trip the RCPs within 13 minutes after the loss of RCP seal cooling results in an RCP seal LOCA. This can have a significant impact on CDF, since under most fire scenarios, long-term cooling is unavailable because the fire causes a direct failure of either the auxiliary component cooling water system or the nuclear service cooling water system.

The HFE with the third highest percentage contribution to CDF (approximately 10 percent) is failure to control AFW given a fire causes the spurious operation of the system (e.g., spurious starting of the AFW pumps). Failure to control AFW flow under these conditions will lead to a loss of secondary-side heat removal.

Mitigating systems are affected by the different fire scenarios modeled in the L3PRA project FPRA, due to individual equipment or trains being directly failed. Since the fire scenarios are dominated by the direct effects of the fire and human errors, random hardware failures are typically not as important as they are for internal event scenarios.

The hardware component failures with the highest percentage contribution to CDF (both approximately 9 percent) are (1) the RCP stage 2 seal failure given all seal cooling is lost and (2) spurious opening of the turbine-driven AFW pump steam inlet valve. The spurious opening of this valve will cause the turbine-driven AFW pump to start, which can lead to overfilling the steam generators and an induced steam line break.

The two component failures with the next highest percentage contribution to CDF (both approximately 5 percent) are failure of the two emergency diesel generators to operate for the mission time. The diesel generators have relatively high importance because multiple fire scenarios cause a LOOP, thereby requiring onsite emergency power to start and operate to provide essential AC power.

All other hardware component failures individually contribute less than 4 percent to total fire CDF.

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- Stacey Hendrickson (formerly of Sandia National Laboratories), Susan Cooper, and Lauren Ning (formerly of the U.S. Nuclear Regulatory Commission) for performing the human reliability analysis and participating in the site visit to the reference plant

- Jose Pires for reviewing the seismic-fire interaction analyses performed by the reference plant

The authors also appreciate Jonathan DeJesus Segarra’s efforts in ensuring that proprietary information from the reference plant was removed, while at the same time maintaining a technically accurate and understandable report and converting the manuscript into the NUREG format.

ABBREVIATIONS AND ACRONYMS

AC	Alternating current
ACCW	Auxiliary component cooling water
ADAMS	Agencywide Documents and Management System
AFW	Auxiliary feedwater
ALWR	Advanced light water reactor
ANS	American Nuclear Society
ARV	Atmospheric relief valves
ASC	Alternate shutdown capability
ASME	American Society of Mechanical Engineers
CAFTA	Computer Aided Fault Tree Analysis
CCDP	Conditional core damage probability
CCU	Containment cooling unit
CCW	Component cooling water
CD	Complete dependence
CDF	Core damage frequency
CST	Condensate storage tank
CVCS	Chemical and volume control system
DC	Direct current
ECCS	Emergency core cooling system
EDG	Emergency diesel generator
EF	Error factor
EPRI	Electric Power Research Institute
ESFAS	Engineered safety features actuating system
FHA	Fire hazard analysis
FIF	Fire ignition frequencies
FLEX	Diverse and Flexible Mitigation Capability
FPIE	Full power internal events
FPRA	Fire probabilistic risk assessment
FV	Fussell-Vesely
F&B	Feed and bleed
GFEHS	Ground fault equivalent hot short
HCLPF	High confidence of a low probability of failure
HD	High dependence
HEP	Human error probability
HFE	Human failure event
HPI	High-pressure injection
HPR	High pressure recirculation
HRA	Human reliability analysis
HRR	Heat release rate

HVAC	Heating, ventilation, and air-conditioning
IE	Initiating event
IPEEE	Individual Plant Examination for External Events
ISINJ	Inadvertent safety injection
ISLOCA	Interfacing system loss-of-coolant accident
JHEP	Joint human error probability
kV	Kilovolt
LD	Low dependence
LERF	Large early release frequency
LMP	Licensing Modernization Project
LOCA	Loss-of-coolant accident
LOOP (or LOSP)	Loss of offsite power
LOOPPC	Plant-centered loss of offsite power
LPI	Low pressure injection
LPR	Low pressure recirculation
LPSD	Low power and shutdown
MCA	Multi-compartment fire analysis
MCR	Main control room
MD	Moderate dependence
MDP	Motor-driven pumps
MFW	Main feedwater
MOV	Motor-operated valves
MSIV	Main steam isolation valves
MSO	Multiple spurious operations
NLWR	Non-light water reactor
NRC	Nuclear Regulatory Commission
NSCW	Nuclear service cooling water
PAU	Physical analysis unit
PORV	Power-operated relief valve
PRA	Probabilistic risk assessment
PSMS	Plant safety monitoring system
PWR	Pressurized-water reactor
RAT	Reserve auxiliary transformers
RCP	Reactor coolant pump
RCS	Reactor coolant system
rcy	Reactor critical year
RHR	Residual heat removal
RIR	Risk Increase Ratio
RRR	Risk Reduction Ratio
RVLIS	Reactor vessel level indicating system
RWST	Refueling water storage tank

SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SAT	Standby auxiliary transformer
SBO	Station blackout
SDP	Shutdown panel
SGTR	Steam generator tube rupture
SI	Safety injection
SLOCA	Small loss-of-coolant accident
SNL	Sandia National Laboratories
SOV	Solenoid-operated valves
SPAR	Standardized plant analysis risk
SSEL	Safe Shutdown Equipment List
TDP	Turbine-driven pump
V	Volt
ZD	Zero dependence
ZOI	Zone of influence

1 INTRODUCTION

The U.S. Nuclear Regulatory Commission (NRC) performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The staff undertook this project in response to Commission direction in the staff requirements memorandum dated September 21, 2011 (Agencywide Documents and Management System [ADAMS] Accession No. ML112640419) resulting from SECY-11-0089, "Options for Proceeding with Future Level 3 Probabilistic Risk Assessment Activities," dated July 7, 2011 (ML11090A039).

As described in SECY-11-0089, the objectives of the L3PRA project are the following:

- Develop a Level 3 PRA, generally based on current state-of-practice methods, tools, and data,[1] that (1) reflects technical advances since the last NRC-sponsored Level 3 PRAs (NRC, 1990), which were completed over 30 years ago, and (2) addresses scope considerations that were not previously considered (e.g., low-power and shutdown risk, multi-unit risk, other radiological sources).

- Extract new insights to enhance regulatory decision making and to help focus limited NRC resources on issues most directly related to the agency's mission to protect public health and safety.

- Enhance PRA staff capability and expertise and improve documentation practices to make PRA information more accessible, retrievable, and understandable.

- Demonstrate technical feasibility and evaluate the realistic cost of developing new Level 3 PRAs.

This report documents the single-unit, reactor at-power, Level 1 PRA for internal fires that supports the L3PRA project. The results provided in this report are for a single unit—a subsequent report in this series addresses multi-unit risk.

Licensee information used in performing the L3PRA project was voluntarily provided based on a licensed, operating nuclear power plant. The information provided reflects the plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained. (For example, the L3PRA does not reflect the current reactor coolant pump shutdown seal design or the potential impact of FLEX strategies.[2]) In addition, the information provided for the reference plant was changed based on additional information, assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models. Finally, it should be noted that the FPRA for the L3PRA project does not account for recent and ongoing work in areas where realism in fire PRA can be improved

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[1] "State-of-practice" methods, tools, and data refer to those that are routinely used by the NRC and industry or have acceptance in the PRA technical community. While the L3PRA project is intended to be a state-of-practice study, note that there are several technical areas within the project scope that necessitated advancements in the state-of-practice (e.g., modeling of multi-unit site risk, modeling of spent fuel in pools or casks, and of human reliability analysis for other than internal events and internal fires).

[2] FLEX refers to the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies. FLEX is intended to maintain long-term core and spent fuel cooling and containment integrity with installed plant equipment that is protected from natural hazards, as well as backup portable onsite equipment. If necessary, similar equipment can be brought from off site.

(e.g., more realistic models of cable damage and functionality, improved heat release rate distributions, better fire modeling techniques, and fire model validation). **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

Since the L3PRA project involves multiple PRA models, each of these models should be considered a "living PRA" until the entire project is complete. It is anticipated that the models and results of the L3PRA project are likely to evolve over time, as other parts of the project are developed, or as other technical issues are identified. As such, the final models and results of the project (which will be documented in a summary report to be published after all technical work for the L3PRA project has been completed) may differ in some ways from the models and results provided in the current report.

The series of reports for the L3PRA project are organized as follows:

Volume 1: Summary (to be published last)

Volume 2: Background, site and plant description, and technical approach

Volume 3: Reactor, at-power, internal event and flood PRA (overview report)

- Volume 3a: Level 1 PRA for internal events
- Volume 3b: Level 1 PRA for internal floods
- Volume 3c: Level 2 PRA for internal events and floods
- Volume 3d: Level 3 PRA for internal events and floods

Volume 4: Reactor, at-power, internal fire and external event PRA (overview report)

- Volume 4a: Level 1 PRA for internal fires
- Volume 4b: Level 1 PRA for seismic events
- Volume 4c: Level 1 PRA for high wind events and other hazards evaluation
- Volume 4d: Level 2 PRA for internal fires and seismic and wind-related events
- Volume 4e: Level 3 PRA for internal fires and seismic and wind-related events

Volume 5: Reactor, low-power and shutdown, internal event PRA (overview report)

- Volume 5a: Level 1 PRA for internal events
- Volume 5b: Level 2 PRA for internal events
- Volume 5c: Level 3 PRA for internal events

Volume 6: Spent fuel pool all hazards PRA (overview report)

- Volume 6a: Level 1 and Level 2 PRA
- Volume 6b: Level 3 PRA

Volume 7: Dry cask storage, all hazards, Level 1, Level 2, and Level 3 PRA

Volume 8: Integrated site risk, all hazards, Level 1, Level 2, and Level 3 PRA

1.1 Notes on Nomenclature

The term or acronym “RP-FPRA” is used for the fire PRA model developed by the reference plant. The acronym “L3-FPRA” is used for the fire PRA model developed by the NRC for the L3PRA project, as discussed and documented in this report.

The phrase “fire scenario” is used both in the RP-FPRA and in the L3-FPRA. Since they refer to different items, the fire scenarios in the RP-FPRA will be referred to as “fire sequences” in this report, whereas the fire scenarios modeled in the L3-FPRA are referred to as “fire scenarios.” Each L3-FPRA fire scenario may contain one or more fire sequences from the RP-FPRA and will be evaluated using a specific event tree. L3-FPRA fire scenarios are discussed in Section 4.

One exception to the above rule involves the phrase “main control room (MCR) abandonment scenario.” Because this phrase is commonly used in the fire PRA community, in this report it is often used interchangeably with “MCR abandonment sequence” in describing the work done in the RP-FPRA.

Lastly, the word “transient” has many definitions depending upon its context. In this report, transient is used in two different contexts. In conventional PRA terminology, it refers to an event that could require a plant trip that might challenge safety systems (NRC, 2013b), that is, transient initiating events (leading to turbine and/or reactor trip). In fire PRA, it also refers to transient combustibles. Both these contexts are used in the L3-FPRA model and this document. To avoid confusion, when the word “transient” is used in this report, the associated context will be identified.

1.2 Summary of Approach

As a part of the L3PRA study, the NRC already constructed a Level 1, at-power, PRA model for internal events (NRC, 2022a and NRC, 2022b) using the Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) software (SAPHIRE, 2017). This Level 1 internal event PRA model is used as the starting point for the internal fire PRA model. The Level 1 internal event PRA model contains all of the potential initiating events, plant response via event tree development, and individual mitigating system response via fault trees. The Level 1 internal event PRA model also contains the success criteria for each of the mitigating systems and the overall plant mission time.

Some areas of the Level 1 internal event PRA model were expanded to account for addition failures that are specific to internal fire events. This expansion involved adding a new event tree for each fire scenario. Additional fault trees were added to account for spurious operations of systems and components that were identified and modeled in the RP-FPRA, along with failure data to account for spurious operation. The fire-related equipment failures are identified in target sets that are associated with each fire scenario. This additional model development is discussed in more detail in Sections 9, 10, and 15.

The work performed to develop the internal fire PRA (L3-FPRA) model is documented in Sections 5 through 19 of this report. The L3-FPRA model is developed and analyzed only for Unit 1.

The RP-FPRA has been peer reviewed and the provided documents address the fire PRA tasks listed in NUREG/CR-6850, “EPRI/NRC-RES Fire PRA Methodology for Nuclear Power Facilities”

(NRC, 2005). The NRC commissioned an independent review of the RP-FPRA. The review was performed to determine the adequacy of the models based on the identified tasks in NUREG/CR-6850. This review was performed by Sandia National Laboratories (SNL). Several items of interest are identified in this review. These items, their impact, and their potential treatment are discussed in the relevant sections of this report.

The next subsection lists all the tasks included in NUREG/CR-6850 (NRC, 2005), and identifies whether the task was directly adopted from the RP-FPRA or was performed, at least in part, by the L3PRA project team.

1.3 Tasks per NUREG/CR-6850 and Report Organization

Later sections of this report and some reference titles refer to fire PRA tasks by number. These task numbers are taken from NUREG/CR-6850. The tasks are listed below with a high level summary describing the extent of work performed by the L3PRA project team. Sections 5 through 20 provide the details of how each of these tasks was addressed in the development of this fire PRA.

TASK 1 – Plant Boundary Definition and Partitioning

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Tasks 1 and 6 was used as necessary in the development of the L3-FPRA. Section 5 discusses the process used to address this task along with the review that was performed by SNL.

TASK 2 – Fire PRA Components Selection

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Task 2 was used as necessary in the development of the L3-FPRA. The target sets based on the component selection were used directly from the RP-FPRA. Section 6 discusses the process used to address this task along with the review that was performed by SNL.

TASK 3 – Fire PRA Cable Selection

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation was used as necessary in the development of the L3-FPRA. The target sets based on the cable selection were used directly from the RP-FPRA. Section 7 discusses the process used to address this task along with the review that was performed by SNL.

TASK 4 – Qualitative Screening

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation was used as necessary in the development of the L3-FPRA. Section 8 discusses the process used to address this task along with the review that was performed by SNL.

TASK 5 – Fire-Induced Risk Model

An adaptation of this task was performed for the L3-FPRA. The task was originally performed by the reference plant and the documentation on Task 5 was used as necessary in the development of the L3-FPRA. The reference plant Computer Aided Fault Tree Analysis (CAFTA) PRA model and corresponding Fire Risk Analysis (FRANX) software files were provided, and these models were used in the development of the L3-FPRA via a scenario mapping process (see Section 9). Section 9 discusses the process used to address this task along with the review that was performed by SNL.

TASK 6 – Fire Ignition Frequencies

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Tasks 1 and 6 was used in the development of the L3-FPRA. The fire ignition frequencies documented in the RP-FPRA were used directly. The fire ignition frequencies were part of the output from the reference plant FRANX files and their use in the L3-FPRA is discussed in Section 10 of this report. Section 10 also discusses the review of this task by SNL. A sensitivity analysis was performed using the fire ignition frequencies from NUREG-2169 (NRC, 2015) and is discussed in Section 19.4.3.8.

TASK 7 – Quantitative Screening

This task was not performed for the L3-FPRA. No information was made available about the quantitative screening process performed for the RP-FPRA. Section 11 discusses the process used to address this task along with the review that was performed by SNL.

TASK 8 – Scoping Fire Modeling

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Tasks 8 and 11 was used as necessary in the development of the L3-FPRA. Section 12 discusses the process used to address this task along with the review that was performed by SNL.

TASK 9 – Detailed Circuit Fire Analysis

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Tasks 3 and 9 was used as necessary in the development of the L3-FPRA. Section 13 discusses the process used to address this task along with the review that was performed by SNL.

TASK 10 – Circuit Failure Mode Likelihood Analysis

This task was not performed for the L3-FPRA. The task was performed by the reference plant and the corresponding documentation on Task 10 was used as necessary in the development of the L3-FPRA. Section 14 discusses the process used to address this task along with the review that was performed by SNL.

TASK 11 – Detailed Fire Modeling

This task was not performed for the L3-FPRA. The detailed fire modeling task was performed by the reference plant and the corresponding documentation on Tasks 8 and 11 was used as reference information in the development of the L3-FPRA. The L3-FPRA used the detailed fire modeling from RP-FPRA as the starting point in the development of the analyzed fire scenarios. The RP-FPRA CAFTA and FRANX detailed model was provided and their use in the L3-FPRA is discussed in Section 15 of this report. Section 15 discusses not only the process used to address this task, but also the review that was performed by SNL.

TASK 12 – Post-Fire Human Reliability Analysis

The L3-FPRA human reliability analysis (HRA) was performed by the L3PRA project team, as discussed in Section 16 and Appendix A of this report. This work is based on the post-fire HRA performed by the reference plant and its corresponding documentation on Task 12. Section 16 also discusses the review that was performed by SNL.

TASK 13 – Seismic-Fire Interactions Assessment

This task was performed by the reference plant and the corresponding documentation on Task 13 was used as necessary in the development of the L3-FPRA. An independent review of the work performed by the reference plant was performed by Division of Engineering staff in the Office of Nuclear Regulatory Research. Section 17 discusses the process used to address this task along with the review that was performed by SNL.

TASK 14 – Fire Risk Quantification

An adaptation of this task was performed for the L3-FPRA. This task was originally performed by the reference plant and the corresponding documentation on Tasks 14 and 15 was used as necessary in the development of the L3-FPRA. The L3-FPRA quantified the fire scenarios developed based on the RP-FPRA model, and in accordance with the mapping approach performed under Task 5. The quantification of these fire scenarios is documented in Section 18 of this report. Section 18 also discusses the review that was performed by SNL.

TASK 15 – Uncertainty and Sensitivity Analyses

The L3PRA project team performed uncertainty and sensitivity analyses on the fire scenarios developed under Task 15. The uncertainty and sensitivity analyses of these fire scenarios are documented in Section 19 of this report. The reference plant documentation on Tasks 14 and 15 was used as necessary in the development of the L3-FPRA. Section 19 also discusses the review that was performed by SNL.

TASK 16 – Fire PRA Documentation

This report documents the work performed to develop the L3-FPRA and its results. Extensive information was adopted from the RP FPRA documentation. Section 20 addresses this task, as well as the review that was performed by SNL.

SUPPORT TASK A Fire Human Failure Event Dependency Analysis (see Appendix A)

2 REFERENCE PLANT FIRE PRA INFORMATION AND USAGE

Development of the L3-FPRA is based on the work performed and documented in the RP-FPRA. Tasks 1 through 11 of NUREG/CR-6850 (NRC, 2005) have been performed and documented for the RP-FPRA in different reports provided by the reference plant. The information in these reports provides the starting point for the L3-FPRA. For example, Task 2 of NUREG/CR-6850, Fire PRA Component Selection Task, establishes and documents the scope of plant components to be modeled in a fire PRA. The associated report for the RP-FPRA documents this task and its results. It identifies plant system components whose proper functioning is relied upon to mitigate the consequences of postulated fire-induced initiating events. The scope also includes the identification of those components whose failure or spurious actuation due to a fire may cause a more complicated or challenging type of initiating event.

Fire compartment and location information is found in the RP-FPRA. The plant boundary information is identified in Appendix 9A of the reference plant final safety analysis report. The reference plant reports document the development of the fire compartments, also referred to as physical analysis units (PAUs), that are referenced in the L3-FPRA.

The fire ignition sources and frequencies are discussed in the RP-FPRA report for Task 6. The fire ignition frequencies as calculated and used in the RP-FPRA are directly used in the L3-FPRA. The use of these frequencies is discussed in later sections of this report.

Cable selection and circuit analysis was performed by the reference plant. This analysis addresses NUREG/CR-6850 Tasks 3 and 9. This information is used directly in the L3-FPRA.

Additional description of the RP-FPRA information and how it was used for the L3-FPRA is provided in Sections 2.1 and 2.2, respectively.

2.1 <u>Reference Plant Fire PRA Information</u>

The RP-FPRA model was peer reviewed. The RP-FPRA reports address all the tasks outlined in NUREG/CR-6850 (NRC, 2005) and provide the information that was used to develop the L3-FPRA model. In addition to the RP-FPRA reports, the L3PRA project team used the reference plant's CAFTA model and the related FRANX output file.

The FRANX output file is the starting set of information to develop the L3-FPRA model. The FRANX output file contains the 3,306 fire sequences that were developed and analyzed by the reference plant. For each fire sequence, the fire ignition frequency, severity factor, fire non suppression probability, scenario conditional core damage probability (CCDP), and resultant core damage frequency (CDF) are provided. The FRANX output file also provides the target sets for each fire scenario. The use of this information to develop the L3-FPRA is discussed in later sections of this report.

The L3-FPRA model heavily leveraged the previous work performed by the reference plant, especially for the following tasks:

- Identification of fire areas (PAUs) to be modeled
- Plant fire walkdowns

- Fire modeling and sequence definitions
- Circuit analysis and spurious actuation modeling
- Data used to quantify scenario frequencies leading to plant trip
- Components damaged (unavailable) due to fire
- Spurious actuation of components

2.2 Reference Plant Fire PRA Usage

The reference plant developed a detailed fire PRA based on the American Society of Mechanical Engineers (ASME)/American Nuclear Society (ANS) PRA Standard (ASME, 2009), and the technical requirements were evaluated according to the guidance provided in NUREG/CR-6850 (NRC 2005). The RP-FPRA model was peer reviewed against Section 4 of the ASME/ANS PRA Standard (ASME, 2009). The peer review document provides detailed information about the findings and suggestions from the peer review team.

The NRC performed a readiness review of the RP-FPRA information, including a limited scope walkdown as an independent evaluation of several dominant fire sequences identified in the RP-FPRA. The readiness review was performed by SNL and the limited scope walkdown was performed by the NRC with support from SNL.

Due to limited access to the reference plant site and other constraints, the L3PRA project team relied heavily on the RP-FPRA for the development of the L3-FPRA. Therefore, the readiness review performed on the RP-FPRA was critically important for determining the acceptability of using the RP-FPRA information for the L3PRA project and identifying potential issues and challenges likely to be faced by the L3PRA project team in its use of this information. The limited scope walkdown also enhanced L3PRA project team confidence in using the RP-FPRA information for this project.

3 DEFINITION OF FIRE SEQUENCES

As discussed in Section 1.1, the word “scenario” is reserved for the fire scenarios defined in the L3-FPRA model; the word “sequence” is used for the scenarios defined in the RP-FPRA model. The fire scenarios in the L3-FPRA model consist of one or more fire sequences from the RP-FPRA model.

The RP-FPRA fire sequence development process addresses Task 8 and Task 11 of NUREG/CR-6850 (NRC, 2005) and the fire scenario selection element of ASME/ANS RA-Sa-2009 (ASME, 2009). The RP-FPRA information discusses the development of fire sequences that are within each of the PAUs (PAUs are also referred to in this report as fire compartments). These PAUs also include control room fire sequences and multi-compartment fire sequences. Each fire sequence has both a calculated raw ignition frequency (i.e., the frequency of the initial fire) and a final frequency that accounts for fire suppression and severity (i.e., incorporates a probability of non-suppression and a fire severity factor).

The L3-FPRA project team accepted the fire sequences as discussed and developed in the RP-FPRA and used them directly as the starting point for the fire scenarios. The next section discusses the details of how the fire sequences from the RP-FPRA are used to develop the L3-FPRA scenarios that were analyzed.

4 L3-FPRA MODEL ASSUMPTIONS AND LIMITATIONS

The L3PRA project level 1 internal event PRA model was the starting point for the L3-FPRA model. Therefore, the assumptions made in the internal event model are also applicable to the L3-FPRA logic. These assumptions are not discussed in this report, but can be found in (NRC, 2022a and NRC, 2022b).

4.1 L3-FPRA Model Assumptions

When developing the L3-FPRA fire scenarios, it was assumed that the fire scenario would cause a reactor trip. Therefore, all fire scenarios, at a minimum, transfer to the general transient event tree for evaluation. The general transient event tree is developed in the Level 1 internal events model to evaluate plant trips. Each fire scenario also transfers to other Level 1 (internal event) event trees, as appropriate, to ensure the complete impact of the fire is evaluated.

Based on information from the RP-FPRA documentation and CAFTA cut sets, an alternate source of offsite power is credited for selected fire scenarios in the L3-FPRA model. To be consistent with the Level 1 internal event PRA, the credit for the alternate source of offsite power is only allowed within the first 2 hours following a loss of offsite power (due to the limited lifetime of the turbine building batteries, which are required for some breaker manipulations needed to restore offsite power to the 4.16 kilovolt [kV] alternating current [AC] safety buses). For some fire scenarios, the specific components failed by the fire may invalidate some of the post-processing rules used to credit the alternate source of offsite power. Therefore, to reestablish consistency in the crediting of the alternate source of offsite power between the internal event and internal fire models, some adjustments were made to the post-processing rules for these fire scenarios. This credit is only given if the necessary equipment/circuits associated with the alternate source of offsite power feed are not damaged due to the fire. Examples of necessary equipment/circuits include the standby auxiliary transformer (SAT) and the 4.16 kV AC breaker on the low voltage side of the SAT.

RP-FPRA fire sequences with the same or very similar CCDPs are assumed to have the same or very similar target sets; therefore, these sequences could potentially be mapped together. During the mapping process, most of the fire sequences that were mapped together were compared to each other to verify that the major target sets were, in fact, the same. Some fire sequences with relatively low CCDPs ($< 3x10^{-3}$) were mapped together even if they involved different fire compartments, as long as both the target sets and the CCDPs were similar.

A detailed cable routing for all systems was not performed during development of the RP-FPRA. Some of the systems that are non-risk significant did not have their cables traced and the RP-FPRA assumed they will be failed for all fires. The systems that are identified as being failed during all fires are main feedwater, instrument air, containment spray, atmospheric dump, and turbine plant cooling water. This assumption was used during the mapping process of the RP-FPRA fire sequences into the L3-FPRA fire scenarios, since the target sets for each L3-FPRA fire scenario used the RP-FPRA flag sets.

The RP-FPRA evaluated main control room abandonment based on visibility and temperature conditions that affect control room habitability due to fires in the control panels. This evaluation, which is consistent with Section 11.5.2.1 of NUREG/CR-6850 (NRC, 2005), was also used in the L3-FPRA model. The modeling approach in the L3-FPRA used the same control room habitability timing, fire-related component failures, and fire suppression information as in the RP-FPRA.

The RP-FPRA documentation identified and discussed 13 assumptions that were used when developing the RP-FPRA fire scenarios. These assumptions deal with cable routing, affected components that are in the zone of influence, CCDP of 1.0 for control room abandonment, etc. These assumptions were also used in the development of target sets and fire scenarios in the L3-FPRA model.

Additional assumptions are identified and characterized in the uncertainty section, in Table 19-2.

4.2 Limitations

A potential limitation of the L3-FPRA model is not individually evaluating all 3,306 fire sequences identified by the reference plant during their fire PRA development. However, 86 percent of the total L3-FPRA CDF either involves mapping the RP-FPRA sequences on a one-to-one basis or involves grouping multiple RP-FPRA sequences together based on having the same CCDP (i.e., only 14 percent of fire CDF is mapped into “residual” bins – discussed further in Section 9). Note, since there are instances where the L3-FPRA model groups similar fire sequences together into a single fire scenario based on the impact of the fire on similar components and fire compartments, some individual fire sequence information is lost, and this could have an impact on the Level 2 PRA modeling.

The RP-FPRA documentation discusses limitations based on using “Generic Fire Modeling Treatment.” This documentation discusses potential limits of the empirical and algebraic models used in calculations and how they can affect the zone of influence, which can determine the potential target sets.

The NRC-commissioned independent review of the RP-FPRA concluded that, in general, the information in the RP-FPRA was suitable for use in the L3-FPRA. However, the review did identify several limitations that might present some level of challenge in developing the L3-FPRA. These limitations include the following:

- Transient fire screening: The RP-FPRA transient fire screening analysis used a relatively low screening heat release rate (69 kW, in comparison to the commonly accepted value of 317 kW). The RP-FPRA also assumed all transient fires were located at floor level rather than some distance above the floor (e.g., at the top of a trash receptacle), which impacts the plume temperature calculations. As a result, it is likely that some potential transient fire scenarios were screened out prematurely.

- Minimal use of fire modeling: For most of the fire sources included in the RP-FPRA risk quantification, the assessment of potential fire damage was based on “visual examination,” rather than fire modeling. Only a relatively small number of higher-risk scenarios were analyzed in additional detail (involving damage and suppression timing analyses). As a result, for most of the quantified fire scenarios, there is no assessment of time to damage and no case-specific assessment of fire suppression (e.g., for cabinet fires, this is all embedded in the severity factor).

- No analysis of alternate shutdown: The RP-FPRA did not credit alternate shutdown capability and assumed a CCDP of 1.0 for all MCR abandonment scenarios. These scenarios were not significant contributors to CDF but represent roughly one-third of the large early release frequency (LERF).

- Fire Frequency: The frequency of cabinet fires (fire frequency Bin 15) may be under-estimated based on the process used in the RP-FPRA. NUREG/CR-6850 (NRC, 2005) states that “well sealed and robustly secured” cabinets should not be included in the cabinet count when apportioning the fire cabinet frequency. However, in the RP-FPRA, some cabinets that were later determined to be well sealed and robustly secured were included in the cabinet count, leading to a dilution in the cabinet fire frequency.

Another limitation involves MCR abandonment, which was only evaluated based on impact to Unit 1. However, this fire scenario has the potential of impacting both Unit 1 and Unit 2. The dual-unit aspects of MCR abandonment will be addressed under the integrated site risk task of the L3PRA project.

Finally, cable routing information is not available for the containment spray system, and it was assumed to fail in all fire scenarios in the RP-FPRA. The exclusion of the containment spray system does not have an impact on the Level 1 CDF results; however, it is important to the Level 2 analysis. In fact, assumed unavailability of containment sprays may be a significant non-conservatism, since the largest effect of spray operation is to increase the probability of combustion failure of the containment. This issue is addressed in more detail in the L3PRA report on the Level 2 PRA for internal fires.

4.3 Model Changes to L3PRA Level 1 PRA Model to Account for Fire Scenarios

The following updates were made to the at-power Level 1 PRA internal event model in order to evaluate the fire scenarios:

1. Added new fire scenario event trees to analyze the identified fire initiating events.

2. Added new fault trees to handle multiple spurious operations of components.

3. Added new conditional probabilities for those components that have fire induced spurious operations.

4. Added new operator actions to account for the new conditions that are created by the fires.

5. Added additional post-processing rules to account for different conditional cut sets created by the fire scenarios along with new operator action dependencies.

6. Added new event tree flag sets to make correct adjustments to the base logic (i.e., change nominal settings of components from FALSE to TRUE). These settings are based on the fire scenario being evaluated. These flag sets will also guarantee failure of impacted components.

These additions and/or model changes to the L3PRA Level 1 PRA model are discussed in Sections 9 and 15.

5 TASK 1 – PLANT BOUNDARY DEFINITION AND PARTITIONING

5.1 Objective of the Task

The objective of Task 1 of NUREG/CR-6850 is to define the physical boundaries and then divide the area within that physical boundary into compartments. At the completion of this task, all the plant fire compartments will be identified for the fire analysis.

5.2 Reference Plant Work Performed on the Task

The reference plant performed the global plant analysis boundary evaluation to identify the fire PAUs. The plant partitioning was performed to define the plant boundaries relevant to the fire PRA and then divide the plant into the PAUs. The PAUs are the starting point of the fire PRA in defining the fire threats to safe shutdown and then can be used for individual bounding analysis.

The RP-FPRA documentation provides a list of criteria used to include/exclude plant locations during the plant boundary and partitioning evaluation. These criteria address whether the fire causes a reactor trip, whether the fire location is connected to primary plant structures, and whether the fire affects equipment required for operation.

The plant boundary and partitioning process evaluated all of the plant structures and they were either screened in or out based on the identified criteria. RP-FPRA documentation provides the list of the reference plant's site structures that were evaluated. Once the plant boundary information was established, the actual partitioning of the plant into the fire compartments was performed. RP-FPRA documentation lists the 443 fire compartments (PAUs) and their boundaries. This information was used directly in the L3-FPRA model by the L3PRA project team because no plant boundary evaluation or partitioning was separately performed.

5.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL performed a review of the reference plant's boundary and partitioning. Based on the review, SNL noted that the evaluation performed was sufficient for the L3PRA project to use the RP-FPRA-identified PAUs.

5.4 L3-FPRA Approach to Address the Task

This task was not performed for the L3-FPRA and relied explicitly on the reference plant's boundary evaluation and the 443 PAUs (fire compartments) as the starting point for this fire PRA model.

6 TASK 2 – FIRE PRA COMPONENT SELECTION

6.1 Objective of the Task

Task 2 provides the basis for identifying components that need to be included into the fire PRA. The identification of the components is also used to develop the corresponding cable information (identification and location). The results of this task are a complete list of equipment to be modeled in the fire PRA and the corresponding cables; both identification and location.

6.2 Reference Plant Work Performed on the Task

The RP-FPRA documentation identifies the following six tasks associated with equipment selection: (1) identification of equipment on the Safe Shutdown Equipment List (SSEL), (2) identification of other equipment to be credited in the fire PRA, (3) identification of unique fire-induced core damage sequences, (4) disposition of PRA basic events, (5) disposition of SSEL components, and (6) definition of surrogates for non-discretely credited components. The reference plant augmented the equipment selection process by considering components that are (1) modeled in the reference plant full power PRA, (2) addressed in the deterministic post-fire safe shutdown analysis, or (3) identified by a review of potential multiple spurious operations (MSOs). The RP-FPRA documentation (1) identifies and uses a process to disposition all of the basic events found in the full power PRA; (2) documents the review of the SSEL, which considers the equipment on the SSEL that needs to be included in the fire PRA depending on differences in deterministic and PRA success criteria or mission time; and (3) documents the review of the interfacing system loss-of-coolant accident (ISLOCA) pathways and each ISLOCA pathway was included in the fire PRA unless it met one of the following three exclusion criteria:

- The path includes flow restrictions that would restrict leakage to a rate below the capacity of normal charging.
- The path is a closed loop inside containment.
- The path contains at least two isolation valves that cannot be impacted by fire.

According to the results of the ISLOCA pathway review, none of the potential ISLOCA pathways were found to require inclusion in the RP-FPRA, even when considering fire-induced multiple spurious operations.

The RP-FPRA documentation also documents the review of the containment penetration pathways and each pathway was included in the fire PRA unless it met one of the following five exclusion criteria:

- The path includes one valve that cannot be impacted by fire.
- The path is a closed loop inside containment (including systems that can be open to the atmosphere, but the radiation release would be small unless there is a primary to secondary side leak).
- The path is water solid, which is considered to be a torturous path for nuclide release.

- The path isolation status is of high awareness and not impacted by fire (e.g., the equipment hatch).

- The path has a diameter of less than 0.75 in.

Finally, the RP-FPRA documents the review of the instrument cues for crediting operator actions and the MSO expert panel review.

6.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL performed a review of the reference plant's component selection. The SNL review looked at the process that the reference plant used to identify components for the fire PRA and deemed it adequate, with a few recommendations. These recommendations are to review ISLOCA pathways and containment pathways due to MSOs, potential expansion of the model to handle Level 2 PRA analyses, and spot check the RP-FPRA model to ensure complete mapping of fire impacts to logically equivalent basic events. The review also identified systems that were assumed to always be impacted (i.e., failed) for all fires; therefore, no cable routing was required. These systems are main feedwater, instrument air, containment spray, turbine plant closed cooling water, and reactor water makeup, as well as steam generator atmospheric relief valves and some 13.8 kV switchgear.

6.4 L3-FPRA Approach to Address the Task

This task was not performed by the L3PRA project for the L3-FPRA. The component selection process used by the reference plant for the RP-PRA is also used in the L3-FPRA.

Regarding the SNL review comments on ISLOCA and containment pathways due to MSOs provided in the previous section, it is noted that the RP-FPRA documentation states that the ISLOCA pathways were re-looked at in light of the "new consensus of the higher probability of fire-induced multiple spurious events." Following this review, the reference plant concluded that "none of the pathways were found to require additional modeling for the Fire PRA."

Also, the SNL review notes the following:

- Based on an industry approach, an expert panel reviewed over 60 potential MSOs in systems for inclusion in the fire PRA model.

- Additions to the fire PRA model were required for some of the identified MSOs, while for others the model already had the necessary structure to evaluate the MSO.

- The completeness of the MSO evaluation was not verified in this readiness review, but the expert panel approach is considered acceptable by NRC/NRR in its review of National Fire Protection Association (NFPA) 805 licensee applications and likely captures the significant MSO cases.

7 TASK 3 – CABLE SELECTION AND CIRCUIT FAILURE ANALYSIS

7.1 Objective of the Task

The objective of Task 3 is to perform cable identification (selection) for all components that were identified in Task 2. The cable identification also identifies the plant routing and location, which is used to evaluate the impact of fires at different locations.

7.2 Reference Plant Work Performed on the Task

The reference plant performed this cable identification concurrently with its detailed circuit failure analysis (Task 9), per industry guidance in NUREG/CR-6850 (NRC, 2005). The RP-FPRA documentation details the methodology, for both circuit analysis and cable routing, to meet the guidance in NUREG/CR-6850, Task 3. The RP-FPRA documentation also describes the database development during the cable selection and circuit analysis and provides (1) the process and criteria used during the cable selection and circuit analysis; (2) the details of the analysis, such as assumptions, analysis criteria, failure modes, and general considerations; and (3) instructions for performing the circuit analysis. Finally, the RP-FPRA documentation identifies each evaluated component, its initial and desired positions, the cables that can affect the component, and the consequences of a fault in these cables.

7.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed this task and assessed that the reference plant's process for performing cable selection was complete and appropriate for at-power operations. However, shutdown equipment and other plant systems that were excluded, for example, those that are assumed to always fail as identified in Task 2, were not part of the cable selection. Therefore, low power and shutdown (LPSD) systems, along with other systems that are not modeled in the at-power Level 1 PRA (e.g., containment spray), will not have any cable routing information to support a LPSD or Level 2 fire PRA.

7.4 L3-FPRA Approach to Address the Task

This task was not performed by the L3PRA project for the L3-FPRA. The cable selection and circuit analysis that was performed by the reference plant for the RP-FPRA was also used in the L3-FPRA.

8 TASK 4 – QUALITATIVE SCREENING

8.1 Objective of the Task

The objective of Task 4 is to perform a qualitative screening of the fire compartments (PAUs) identified in Task 1. This task is not intended to develop risk values for each fire compartment, but to identify fire compartments that are expected to have low risk or nonexistent impact compared to others.

8.2 Reference Plant Work Performed on the Task

The reference plant performed this task in conjunction with Task 1. The screening criteria that were used to screen in or out plant boundaries are listed in the RP-FPRA documentation (as previously discussed in Section 5.2 of this [L3PRA project] report). The RP-FPRA documentation provides a description of those plant boundaries that were screened out from further analysis.

8.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed this task and assessed that the reference plant's qualitative screening for at-power operation was complete and appropriate for use in the fire PRA being developed by the L3PRA project.

8.4 L3-FPRA Approach to Address the Task

This task was not performed by the L3PRA project for the L3-FPRA. The cable selection and circuit analysis that were performed by the reference plant for the RP-FPRA were also used in the L3-FPRA.

9 TASK 5 – FIRE-INDUCED RISK MODEL

9.1 Objective of the Task

Task 5 is structured to provide procedures on the development of a fire PRA model. This task identifies the methods and processes to take an internal event PRA model and implement both temporary and permanent changes in order to develop a fire risk model that will provide CDF, CCDP, LERF, and conditional large early release probability.

9.2 Reference Plant Work Performed on the Task

The reference plant developed the RP-FPRA starting with their full power internal event PRA model. The RP-FPRA documentation explains any model structural changes that were required to take the internal event full power PRA model and develop the fire PRA model and discusses what MSO, instrumentation, LERF, and other additional model changes were performed.

The RP-FPRA documentation also discusses changes that were needed to specific events in the development of the RP-FPRA model. The RP-FPRA documentation highlights the new flag sets added; the two operator recovery actions credited in the internal event PRA model (offsite power and nuclear service cooling water [NSCW] valves), neither of which is credited in the fire PRA; treatment of Information Notice (IN) 92-18[3]; operator action adjustment and new actions added; initiating event frequencies; modified basic event probabilities; and events removed from the fire PRA. The RP-FPRA development took the 443 fire compartments (PAUs) identified in Task 1 and created 3,306 fire sequences based on the fire modeling. These 3,306 fire sequences were analyzed using the flag sets and other modifications to the at-power PRA model.

The RP-FPRA documentation also discusses how the FRANX software was used to analyze the fire PRA model and explains how the cabling and fire compartments were applied to the internal event model to correctly handle the conditions of each fire compartment. The RP-FPRA documentation highlights the conditional probability for spurious operation of valves and how their probabilities changed between different data sources. The FRANX software is used to apply the developed flag sets to the internal event PRA model in order to obtain the overall fire CDF. To accomplish this, the analyst creates a specific flag set that identifies the condition that a specific fire sequence can cause. This condition includes what components are failed, what components can spuriously operate, and/or what operator actions are required. There are a total of 3,306 specific flag sets created to evaluate the 3,306 identified fire sequences. The FRANX software streamlines the process of applying these flag sets to the internal event PRA model for analysis.

9.3 SNL Review of the Reference Plants Approach to Address the Task

SNL reviewed this task and assessed that the model was appropriate given the model enhancements and error corrections that were made based on the fire PRA peer review. SNL

[3] IN 92-18 refers to Loss of Remote Shutdown Capability during a Control Room Fire. This information notice addresses the potential for a control room fire to cause electrical short circuits between normally energized conductors and conductors associated with control circuitry.

did not review the actual PRA model and relied on the RP-FPRA documentation, along with information provided by the ASME standard PRA peer review that was performed.

9.4 L3-FPRA Approach to Address the Task

The L3PRA project team performed an adaption of this process in the development of the fire PRA model. The L3-FPRA started with the L3PRA internal event PRA model and then overlayed the different fire scenarios based on the RP-FPRA information. The reference plant CAFTA PRA model and the corresponding FRANX software files were provided, and these models were used in the development of the L3-FPRA via fire scenario mapping. The mapping of the RP-FPRA fire sequences into the L3-FPRA fire scenarios is a multi-step process, which is summarized in Section 9.4.1. The fire scenario definitions are described in Section 9.4.2, and Section 9.4.3 describes the development of the L3-FPRA logic model.

9.4.1 Mapping of Fire Sequences into Fire Scenarios

The reference plant developed an NFPA 805 compliant internal fire PRA. The corresponding RP-FPRA documentation addresses the tasks outlined in NUREG/CR-6850 (NRC, 2005). This documentation, along with the 443 fire compartments (PAUs) and 3,306 fire sequences analyzed by the reference plant, were used as references and served as the starting point in the development of the L3-FPRA.

The RP-FPRA information on fire sequences and affected components was used directly in the L3-FPRA; no additional walkdowns or verification were performed to support the L3-FPRA model development. Plant walkdowns were performed by the L3PRA project team as part of the review of the RP-FPRA; however, the information from this review did not impact the L3-FPRA base case model development.

The RP-FPRA fire sequences were used to create the set of internal fire scenarios that were placed in the L3-FPRA model. The RP-FPRA fire sequences were mapped into a smaller subset (discussed below) to be analyzed, as opposed to analyzing all 3,306 fire sequences individually in the L3-FPRA. This mapping (grouping) approach was used to group similar fire sequences together in order reduce the number of fire sequences to a manageable set of fire scenarios to be placed in the L3-FPRA model. The mapping process involved identifying the fire sequences defined and quantified in each dominant PAU from the RP-FPRA and mapping them into a set of fire scenarios to be modeled in the L3-FPRA. The fire scenario definition process and summary results are discussed in this report.

The process of mapping the RP-FPRA fire sequences to L3-FPRA fire scenarios involved the following steps:

- All 3,306 RP-FPRA fire sequences were ordered by decreasing CDF and all fire sequences with a CDF below the 10^{-12}/ry truncation level were eliminated (these sequences had a reported CDF of 0.0).
- The remaining 2,481 RP-FPRA fire sequences were ordered by decreasing CCDP.
- All fire sequences with a CCDP ≥ $3x10^{-3}$ were reviewed and any sequences that (1) occur in the same fire compartment, (2) have the same CCDP, and (3) impact the same safety equipment, were mapped (grouped) together into a single L3-FPRA fire scenario.

- All RP-FPRA fire sequences with a CCDP ≥ $3x10^{-3}$ but that could not be grouped based on the above criteria, were mapped on a one-to-one basis into a L3-FPRA fire scenario.

- The remaining RP-FPRA fire sequences (i.e., those with a CCDP < $3x10^{-3}$) were reviewed to look for patterns in the sequence CCDPs to enable further mapping (grouping) of multiple RP-FPRA fire sequences into single L3-FPRA fire scenarios.

The above process resulted in 566 RP-FPRA fire sequences being mapped into 162 L3-FPRA fire scenarios. These 162 L3-FPRA fire scenarios account for 86 percent of the total L3-FPRA fire CDF. Of the 162 L3-FPRA fire scenarios, 102 are mapped directly on a one-to-one basis and contribute 48 percent to the total fire CDF. The other 60 L3-FPRA fire scenarios, which involve multiple RP-FPRA fire sequences grouped together per the above process, contribute 38 percent to the total fire CDF.

The remaining 1,915 residual fire sequences were grouped based on similar CCDP, as well as any unique nature of the fire sequences. These residual fire sequences were mapped into 48 different fire scenarios (which collectively contribute 14 percent to the total fire CDF). These fire scenarios are labeled with “RR” to recognize that they go across fire compartments. Care must be taken in grouping the residual fire scenarios (1) because of the potential to overestimate the resultant CDF and (2) to avoid grouping fire sequences that impact “A” train mitigating equipment with fire sequences that impact “B” train mitigating equipment. As such, this process of reviewing and mapping the residual fire sequences was performed through many iterations until all residual fire sequences were assigned to a L3-FPRA fire scenario.

The set of impacted equipment used to evaluate each residual fire scenario is obtained from the RP-FPRA fire sequence with the highest CCDP that is mapped to that L3-FPRA fire scenario. Note, while it would be more appropriate and useful to map scenarios based on actual plant impact (in terms of equipment affected by the fire) rather than CCDP, given the large number of quantified fire sequences in the RP-FPRA, and the large number of potential impacts per sequence, this was not practical as part of the L3PRA project. However, in limited situations, the actual plant impact for selected residual fire sequences was investigated to confirm that the impacted equipment was essentially the same. Also, detailed review of the final results (as discussed in Section 18.4.2) helped confirm that grouping sequences based solely on CCDP did not have a significant impact on the fire quantification results.

An internal review was performed to ensure the L3-FPRA fire scenarios were mapped consistent with this process, and that the final grouping is reasonable. Note, since the mapping process involves a lot of subjective judgment, other independent applications of this (or a similar) process would likely lead to different mapping results and a different number of L3-FPRA fire scenarios. Nonetheless, based on a detailed review of the final results and review of the impacted equipment for selected fire scenarios, as discussed above, this mapping process is believed to be sound.

In defining the fire scenarios, care was taken to conserve the total fire compartment (PAU) sequence initiating event frequency and, to the extent practical, the total fire compartment CDF from the RP-FPRA. However, for several reasons (e.g., differences in internal event PRA modeling assumptions), it was not expected that the L3-FPRA fire CDF would match the RP-FPRA fire CDF. The differences in the fire compartment CDFs between the L3-FPRA and the RP-FPRA are examined in Section 18, as part of the model review effort.

The L3-FPRA model is expected to provide a good estimate of the total plant CDF from internal fires and provide a good estimate of major contributors to fire risk, in terms of scenarios or major structures, systems, and components (SSCs).

The fire scenarios defined for placement into the L3-FPRA model from different fire compartments were collected in a single table that established the starting point for placing the fire scenarios into the L3-FPRA model.

Some examples of cases where multiple RP FPRA sequences are combined into a single L3-FPRA scenario are provided below:

- 480 V AC Switchgear Room – Hot gas layer sequences from all ignition sources are gathered into one fire scenario and the scenario frequencies are summed. The CCDP for these sequences is the same; thus, there is no numerical approximation made.

- Switchgear room battery charger fire sequences are gathered into one scenario and the scenario frequencies are summed. The CCDP is the same for these sequences.

- The transformer yard sequences are gathered into one scenario and the frequencies are summed. All of these fires cause a loss of offsite power (LOOP).

9.4.2 Fire Scenario Definitions

The RP-FPRA plant fire model has been defined in terms of internal fire sequences. These sequences can be examined individually and/or as one fire compartment at a time; a fire compartment with a high contribution to fire CDF, such as 1092-J9, will be used as an example. This example fire compartment is the Unit 1 Control Building Level A Train B 4.16 kV AC Switchgear Room A050. The sequences defined and quantified in the RP-FPRA for this compartment are listed in Table 9-1.

The sequences in Table 9-1 are sorted by their CCDP. It is assumed that the sequences with the same CCDP have the same or similar target sets, regardless of the ignition source and fire propagation (this is verified by reviewing the FRANX flag set files provided by the reference plant). The first eight sequences defined in Table 9-1 are for the electric cabinet fires that have the most consequence (i.e., CCDP between 4.23×10^{-2} and 4.27×10^{-2}). These eight sequences are mapped into a single L3-FPRA scenario, named 1-IE-FRI-1092-J9_C104, with a combined sequence frequency of 4.51×10^{-5}/rcy (see Table 9-2). Based on information provided and the target sets, these eight fire sequences are essentially the same. The reason for the slightly different CCDP is the truncation value used when these fire sequences were evaluated in the RP-FPRA.

Similarly, the 20 fire sequences with a CCDP of between 2.09×10^{-3} and 2.23×10^{-3} are mapped into a single L3-FPRA scenario named 1-IE-FRI-1092-J9_C113.

Each L3-FPRA fire scenario may be comprised of one or more fire sequence(s) with different ignition sources; however, a single target set is used for the fire scenario. The target set that is used for the fire scenario is based on reviewing the target sets from the individual fire sequences that comprise it, to make sure the representative fire scenario encompasses the collective impacts of all of the included fire sequences. The CCDP cells that are highlighted with the same color in Table 9-1 have been grouped into a single fire scenario and modeled in the

L3-FPRA. The remaining fire sequences have been modeled in one of two ways: (1) a single-sequence fire scenario (the two fire sequences that have their CDF highlighted, that is, sequences 1092-J9_D0 and 1092-J9_E0) or (2) a residual fire scenario (the remaining unhighlighted fire sequences).

A single-sequence fire scenario takes a fire sequence directly from the RP-FPRA and analyzes it individually (i.e., there is a one-to-one mapping of the RP-FPRA fire sequence to the L3-FPRA fire scenario). The only RP-FPRA fire sequences that are mapped on a one-to-one basis to L3-FPRA fire scenarios are those that contribute significantly to total fire CDF. This is to limit the overall number of fire scenarios that need to be analyzed in the L3-FPRA, while ensuring that these highly dominant (important) fire sequences are captured in the L3-FPRA.

Residual fire scenarios involve grouping multiple RP-FPRA fire sequences into a single L3-FPRA fire scenario regardless of the associated fire compartments. This grouping is used as a means to capture the fire sequences that are not dominant contributors to CDF and CCDP. These residual fire scenarios are developed to ensure that the total RP-FPRA fire initiating event frequency is preserved, as well as the total RP-FPRA fire CDF. These fire sequences are grouped together into single fire scenarios based on similar CDF, CCDP, and initiating event frequency.

Each mapped L3-FPRA fire scenario becomes the initiating event for a standard event tree logic structure. The mapped fire scenarios have specific target sets that are identified from the RP-FPRA. These target sets (components and cabling that are failed or spuriously affected) get mapped into the event tree (or supporting fault trees). The event tree logic and the conditional plant response fault tree logic are discussed in the next section.

Table 9-1 RP-FPRA Sequences 1092-J9 to be Mapped into Differenct L3-FPRA Fire Scenarios

Sequence	Sequence Description	IGF (/rcy)	Severity Factor	NSP	Sequence Frequency (/rcy)	CCDP	CDF (/rcy)
1092-J9_C104	4.16 kV AC Switchgear 1BA03 Cub. 04 Fire	3.98E-05	0.19	1.00	7.56E-06	4.27E-02	3.23E-07
1092-J9_C105	4.16 kV AC Switchgear 1BA03 Cub. 05 Fire	3.98E-05	0.19	1.00	7.56E-06	4.27E-02	3.23E-07
1092-J9_C106	4.16 kV AC Switchgear 1BA03 Cub. 06 Fire	3.98E-05	0.19	1.00	7.56E-06	4.27E-02	3.23E-07
1092-J9_C107	4.16 kV AC Switchgear 1BA03 Cub. 07 Fire	3.98E-05	0.19	1.00	7.56E-06	4.27E-02	3.23E-07
1092-J9_C004	4.16 kV AC Switchgear 1BA03 Cub. 04 HEAF	3.71E-06	1.00	1.00	3.71E-06	4.23E-02	1.57E-07
1092-J9_C005	4.16 kV AC Switchgear 1BA03 Cub. 05 HEAF	3.71E-06	1.00	1.00	3.71E-06	4.23E-02	1.57E-07
1092-J9_C006	4.16 kV AC Switchgear 1BA03 Cub. 06 HEAF	3.71E-06	1.00	1.00	3.71E-06	4.23E-02	1.57E-07
1092-J9_C007	4.16 kV AC Switchgear 1BA03 Cub. 07 HEAF	3.71E-06	1.00	1.00	3.71E-06	4.23E-02	1.57E-07
1092-J9_C3	4.16 kV AC Switchgear 1BA03	9.16E-04	1.00	0.00	2.17E-07	4.05E-02	8.78E-09
1092-J9_E2	Train B Safety Features Sequencer Cabinet 1-1821-U	1.99E-04	1.00	0.00	1.74E-07	4.02E-02	6.99E-09
1092-J9_D2	Cabinet 1BCPAR9 Fire	3.98E-05	1.00	0.00	3.48E-08	3.66E-02	1.27E-09
1092-J9_C204	4.16 kV AC Switchgear 1BA03 Cub. 04 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	7.67E-03	2.47E-07
1092-J9_C205	4.16 kV AC Switchgear 1BA03 Cub. 05 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	7.67E-03	2.47E-07
1092-J9_C206	4.16 kV AC Switchgear 1BA03 Cub. 06 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	7.67E-03	2.47E-07
1092-J9_D0	Cabinet 1BCPAR9 Fire	3.98E-05	0.58	1.00	2.32E-05	2.83E-03	6.58E-08
1092-J9_E0	Train B Safety Features Sequencer Cabinet 1-1821-U	1.99E-04	0.58	1.00	1.16E-04	2.60E-03	3.02E-07
1092-J9_C100	4.16 kV AC Switchgear 1BA03 Cub. 00 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C101	4.16 kV AC Switchgear 1BA03 Cub. 01 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C102	4.16 kV AC Switchgear 1BA03 Cub. 02 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C103	4.16 kV AC Switchgear 1BA03 Cub. 03 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C108	4.16 kV AC Switchgear 1BA03 Cub. 08 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C109	4.16 kV AC Switchgear 1BA03 Cub. 09 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C110	4.16 kV AC Switchgear 1BA03 Cub. 10 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C111	4.16 kV AC Switchgear 1BA03 Cub. 11 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C112	4.16 kV AC Switchgear 1BA03 Cub. 12 Fire	3.98E-05	0.19	1.00	7.56E-06	2.48E-03	1.88E-08
1092-J9_C000	4.16 kV AC Switchgear 1BA03 Cub. 00 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C001	4.16 kV AC Switchgear 1BA03 Cub. 01 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C002	4.16 kV AC Switchgear 1BA03 Cub. 02 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C003	4.16 kV AC Switchgear 1BA03 Cub. 03 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C008	4.16 kV AC Switchgear 1BA03 Cub. 08 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C009	4.16 kV AC Switchgear 1BA03 Cub. 09 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09

Table 9-1 RP-FPRA Sequences 1092-J9 to be Mapped into Differenct L3-FPRA Fire Scenarios

Sequence	Sequence Description	IGF (/rcy)	Severity Factor	NSP	Sequence Frequency (/rcy)	CCDP	CDF (/rcy)
1092-J9_C010	4.16 kV AC Switchgear 1BA03 Cub. 10 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C011	4.16 kV AC Switchgear 1BA03 Cub. 11 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C012	4.16 kV AC Switchgear 1BA03 Cub. 12 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.28E-03	8.44E-09
1092-J9_C113	4.16 kV AC Switchgear 1BA03 Cub. 13 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C114	4.16 kV AC Switchgear 1BA03 Cub. 14 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C115	4.16 kV AC Switchgear 1BA03 Cub. 15 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C116	4.16 kV AC Switchgear 1BA03 Cub. 16 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C117	4.16 kV AC Switchgear 1BA03 Cub. 17 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C118	4.16 kV AC Switchgear 1BA03 Cub. 18 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C119	4.16 kV AC Switchgear 1BA03 Cub. 19 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C120	4.16 kV AC Switchgear 1BA03 Cub. 20 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C121	4.16 kV AC Switchgear 1BA03 Cub. 21 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C122	4.16 kV AC Switchgear 1BA03 Cub. 22 Fire	3.98E-05	0.19	1.00	7.56E-06	2.23E-03	1.69E-08
1092-J9_C013	4.16 kV AC Switchgear 1BA03 Cub. 13 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C014	4.16 kV AC Switchgear 1BA03 Cub. 14 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C015	4.16 kV AC Switchgear 1BA03 Cub. 15 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C016	4.16 kV AC Switchgear 1BA03 Cub. 16 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C017	4.16 kV AC Switchgear 1BA03 Cub. 17 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C018	4.16 kV AC Switchgear 1BA03 Cub. 18 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C019	4.16 kV AC Switchgear 1BA03 Cub. 19 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C020	4.16 kV AC Switchgear 1BA03 Cub. 20 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C021	4.16 kV AC Switchgear 1BA03 Cub. 21 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_C022	4.16 kV AC Switchgear 1BA03 Cub. 22 HEAF	3.71E-06	1.00	1.00	3.71E-06	2.09E-03	7.74E-09
1092-J9_TR03	TRANSIENT AT SOUTH WALL	4.33E-06	1.00	1.00	4.33E-06	6.63E-04	2.87E-09
1092-J9_E1	Train B Safety Features Sequencer Cabinet 1-1821-U	1.99E-04	0.42	1.00	8.28E-05	1.95E-04	1.61E-08
1092-J9_C221	4.16 kV AC Switchgear 1BA03 Cub. 21 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.94E-04	6.24E-09
1092-J9_C207	4.16 kV AC Switchgear 1BA03 Cub. 07 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C208	4.16 kV AC Switchgear 1BA03 Cub. 08 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.22E-09
1092-J9_C209	4.16 kV AC Switchgear 1BA03 Cub. 09 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.22E-09

Table 9-1 RP-FPRA Sequences 1092-J9 to be Mapped into Differenct L3-FPRA Fire Scenarios

Sequence	Sequence Description	IGF (/rcy)	Severity Factor	NSP	Sequence Frequency (/rcy)	CCDP	CDF (/rcy)
1092-J9_C210	4.16 kV AC Switchgear 1BA03 Cub. 10 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.22E-09
1092-J9_C211	4.16 kV AC Switchgear 1BA03 Cub. 11 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C212	4.16 kV AC Switchgear 1BA03 Cub. 12 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C213	4.16 kV AC Switchgear 1BA03 Cub. 13 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C214	4.16 kV AC Switchgear 1BA03 Cub. 14 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C215	4.16 kV AC Switchgear 1BA03 Cub. 15 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C216	4.16 kV AC Switchgear 1BA03 Cub. 16 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C217	4.16 kV AC Switchgear 1BA03 Cub. 17 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C222	4.16 kV AC Switchgear 1BA03 Cub. 22 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.93E-04	6.24E-09
1092-J9_C203	4.16 kV AC Switchgear 1BA03 Cub. 03 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.92E-04	6.19E-09
1092-J9_C218	4.16 kV AC Switchgear 1BA03 Cub. 18 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.92E-04	6.19E-09
1092-J9_C219	4.16 kV AC Switchgear 1BA03 Cub. 19 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.92E-04	6.19E-09
1092-J9_C220	4.16 kV AC Switchgear 1BA03 Cub. 20 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.92E-04	6.19E-09
1092-J9_C200	4.16 kV AC Switchgear 1BA03 Cub. 00 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.91E-04	6.18E-09
1092-J9_C201	4.16 kV AC Switchgear 1BA03 Cub. 01 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.91E-04	6.18E-09
1092-J9_C202	4.16 kV AC Switchgear 1BA03 Cub. 02 Fire - No Target Damage	3.98E-05	0.81	1.00	3.22E-05	1.91E-04	6.18E-09
1092-J9_D1	Cabinet 1BCPAR9 Fire - No Target Damage	3.98E-05	0.42	1.00	1.66E-05	7.42E-06	1.23E-10
1092-J9_TR02	TRANSIENT AT NORTH WALL	4.33E-06	1.00	1.00	4.33E-06	2.35E-07	1.02E-12

Table 9-1 RP-FPRA Sequences 1092-J9 to be Mapped into Differenct L3-FPRA Fire Scenarios

Sequence	Sequence Description	IGF (/rcy)	Severity Factor	NSP	Sequence Frequency (/rcy)	CCDP	CDF (/rcy)
1092-J9_JB1	Junction Box	3.38E-06	1.00	1.00	3.38E-06	0.00E+00	0.00E+00
1092-J9_TR01	TRANSIENT AT SW CORNER	1.84E-06	1.00	1.00	1.84E-06	0.00E+00	0.00E+00
1092-J9_TR04	TRANSIENT AT EAST WALL	1.84E-06	1.00	1.00	1.84E-06	0.00E+00	0.00E+00
		3.56E-03			1.26E-03		3.68E-06

Notes:
/rcy – per reactor critical year
IGF – fire sequence ignition frequency
Severity Factor – conditional probability that given a fire has occurred, it will result in target damage
NSP – non-suppression [probability of non-suppression (automatic and/or manual)]
Sequence Frequency – the sequence frequency is the initiating event frequency (IGF * Severity Factor * non-suppression)
CCDP – conditional core damage probability, which is obtained by dividing the sequence CDF by the sequence frequency [initiating event frequency]
CDF – core damage frequency, which is obtained through quantification of the fire sequence using the sequence frequency, fire damage vector, and plant response (PRA) model

Table 9-2 RP-FPRA Sequences Mapped into L3-FPRA 1-IE-FRI-1092-J9_C104 Fire Scenario

RP-FPRA Sequence	Sequence Description	IGF (/rcy)	Severity Factor	NSP	Sequence Frequency (/rcy)	CCDP	CDF (/rcy)
1092-J9_C004	4.16 kV AC Switchgear 1BA03 Cub. 04 HEAF	3.71E-06	1	1	3.71E-06	4.23E-02	1.57E-07
1092-J9_C005	4.16 kV AC Switchgear 1BA03 Cub. 05 HEAF	3.71E-06	1	1	3.71E-06	4.23E-02	1.57E-07
1092-J9_C006	4.16 kV AC Switchgear 1BA03 Cub. 06 HEAF	3.71E-06	1	1	3.71E-06	4.23E-02	1.57E-07
1092-J9_C007	4.16 kV AC Switchgear 1BA03 Cub. 07 HEAF	3.71E-06	1	1	3.71E-06	4.23E-02	1.57E-07
	Partial Sum =				1.48E-05		
1092-J9_C104	4.16 kV AC Switchgear 1BA03 Cub. 04 Fire	3.98E-05	0.19	1	7.56E-06	4.27E-02	3.23E-07
1092-J9_C105	4.16 kV AC Switchgear 1BA03 Cub. 05 Fire	3.98E-05	0.19	1	7.56E-06	4.27E-02	3.23E-07
1092-J9_C106	4.16 kV AC Switchgear 1BA03 Cub. 06 Fire	3.98E-05	0.19	1	7.56E-06	4.27E-02	3.23E-07
1092-J9_C107	4.16 kV AC Switchgear 1BA03 Cub. 07 Fire	3.98E-05	0.19	1	7.56E-06	4.27E-02	3.23E-07
	Partial Sum =				3.02E-05		
	Grand total of sequence frequencies to be input as the frequency of SAPHIRE scenario 1-IE-FRI-1092-J9_C104				4.51E-05		
Note: Refer to Table 9-1 for column headings definitions.							

9.4.3 Development of the L3-FPRA Model

This section describes the development of the L3-FPRA logic model. The general structure is described in Section 9.4.3.1. Details of the event tree and fault tree models are provided in Sections 9.4.3.2 and 9.4.3.3, respectively.

9.4.3.1 L3-FPRA Model Structure

The L3-FPRA model starts with the L3PRA Level 1 PRA model for internal events and then modifies it by adding new logic in order to evaluate the different fire scenarios. The L3-FPRA model uses the reference plant fire reports along with the RP-FPRA CAFTA and FRANX model for the fire scenario model development. The modifications include adding new event tree logic, fault tree logic, and basic events. The new event trees and fault trees are discussed in the following sections, and the new event tree initiating event frequencies are listed in Table 15-2.

The L3-FPRA model was generated and quantified using SAPHIRE (SAPHIRE, 2017). The SAPHIRE software incorporates scenario-specific event trees to generate accident sequences. These accident sequences are then solved to determine the minimal groups of components (cut sets) that will lead to core damage. These minimal groups of components are then quantified to determine the overall CDF.

9.4.3.2 Event Tree Models

A standard event tree structure was developed in order to analyze each of the identified fire scenarios (a total of 210 different fire scenarios). This event tree logic structure was created such that it captures all of the impacts due to the fire scenario as identified in the RP-FPRA fire sequences.

All fires that are analyzed will cause a reactor trip; however, the fire may also result in other conditions that require different plant responses; for example, a consequential LOOP or LOCA, or an inadvertent safety injection signal. One way to handle this is to review the dominant impact of each fire scenario (via cut set review of the RP-FPRA) and have the fire scenario transfer to just one particular event tree from the Level 1 internal event PRA. However, this has the potential of missing the full impact of the fire scenario on the plant. Therefore, the process that is used in the L3-FPRA is to transfer each fire scenario to eight different event trees from the Level 1 internal event PRA listed in Table 9-3 (see Section 9.4.3.3.2 for a discussion of why these specific eight event trees were chosen). This approach allows the full impact of the fire and the associated plant response to determine the final results. Figure 9-1 shows the standard structure for each of the 210 fire scenarios. The fire scenario event tree questions a conditional fault tree with eight sub-fault trees (i.e., those listed in Table 9-3) prior to the transfer. The conditional fault tree is used to determine the condition of the plant’s mitigating equipment (e.g., failed or spuriously operated) and condition of the plant (e.g., whether a LOOP has occurred).

Table 9-3 Fire Scenario/Fault Tree Transfers to Level 1 Internal event PRA Event Trees

Fire Scenario/ Fault Tree	Fire Scenario/Fault Tree Description	Transfer to Level 1 Internal Events PRA Event Tree
1-FIRE-RTRIP	Fire-induced reactor trip	1-FPI-OTRANS
1-FIRE-ISINJ	Fire-induced inadvertent safety injection	1-FPI-ISINJ
1-FIRE-LOSINJ	Fire-induced loss of seal injection	1-FPI-LOSINJ
1-FIRE-SLOCA	Fire-induced small LOCA	1-FPI-SLOCA
1-FIRE-MLOCA	Fire-induced medium LOCA	1-FPI-MLOCA
1-FIRE-CSLOCA	Fire-induced consequential LOCA	1-FPI-CSLOCA
1-FIRE-SSBI	Fire-induced secondary side break upstream of MSIVs	1-FPI-SSBI
1-FIRE-LOSP	Fire-induced LOOP (plant-centered)	1-FPI-LOOPPC

The eight fault tree logic models are described in more detail in Section 9.4.3.3. The remainder of this section addresses:

- Scenario event trees and transfers (Section 9.4.3.2.1)
- Event tree post-processing rules in the L3-FPRA (Section 9.4.3.2.2)
- Example target set/flag file implementation (Section 9.4.3.2.3)
- Offsite power and EDG recovery assumptions (Section 9.4.3.2.4)

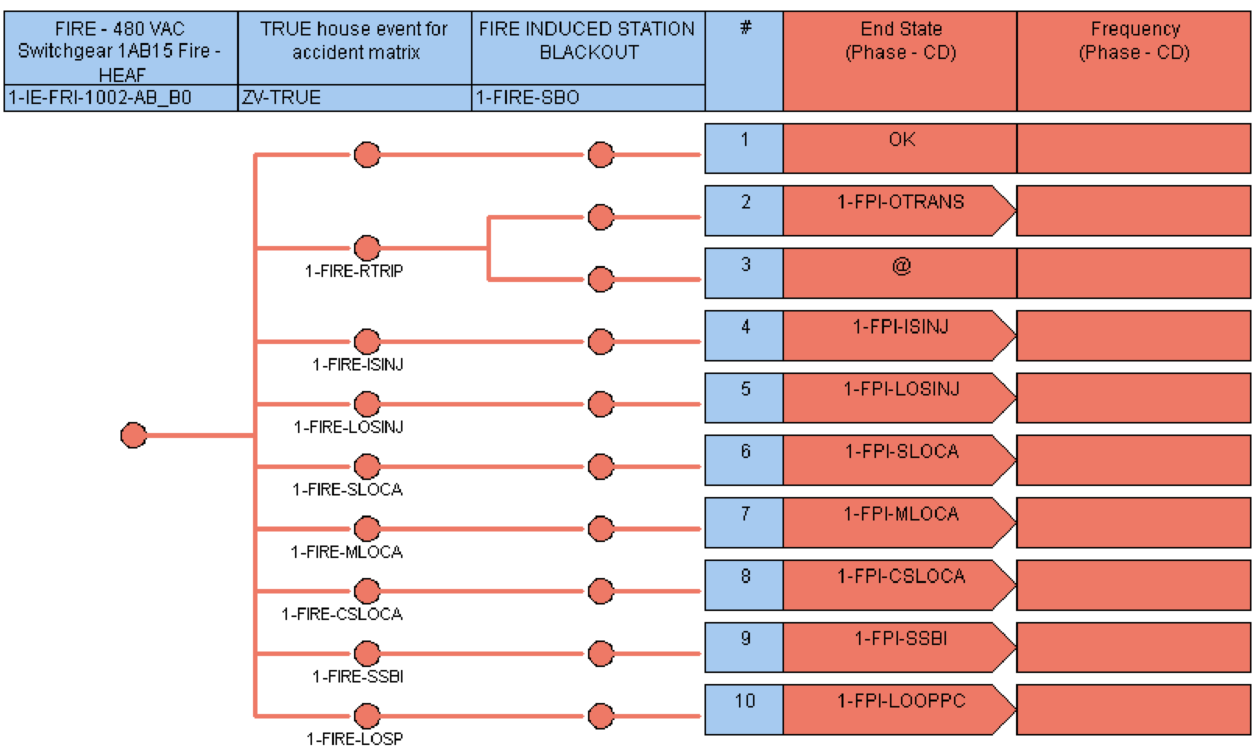

Figure 9-1 Event Tree Used to Map Fire Impacts

9.4.3.2.1 Scenario Event Trees and Transfers

A fire in the plant can lead to a number of functional impacts. These functional impacts can be mapped into the already defined event trees in the L3PRA Level 1 PRA model for internal events. This mapping process is performed using a combination of conditional fault trees (described in Section 9.4.3.3.2) and the existing Level 1 internal event trees to capture the full impact of each fire scenario. Figure 9-1 shows the general structure of the fire scenario event tree that is used to propagate the fire impacts to the existing L3PRA Level 1 internal event trees. The standard fire event tree contains three top events. The first top event represents the fire scenario being evaluated and it contains the initiating event frequency. The second top (ZV-TRUE) is modified by the fault tree listed at each branch node to set up the impact of the fire on the plant (i.e., which mitigating components are failed and/or spuriously change state based on the specific fire). These conditional plant response fault trees are discussed further in Section 9.4.3.3.2.

The last top event, 1-FIRE-SBO fault tree, is modeled only for the general transient event tree, and is designed to account for uncomplicated fire-induced reactor trips (i.e., fires that do not cause any spurious operations, induced actuation signals, or other impacts on plant response capability). The 1-FIRE-SBO fault tree can be viewed as a means to parse uncomplicated fires into either a fire-induced reactor trip with offsite AC power available or a fire-induced reactor trip that involves a LOOP. If the fire causes just a reactor trip and offsite power is available, then the mitigating systems are dependent upon offsite AC power and (based on the 1-FIRE-SBO logic) transfer through the general transient event tree. However, if the fire causes a reactor trip and a LOOP (or a LOOP occurs subsequent to the reactor trip, which is modeled as a plant-centered LOOP [LOOPPC]), then the associated cut sets are assigned to the "@" end state[4] instead of the core damage end state, which eliminates them from the core damage frequency calculation. Therefore, such cut sets only go through the LOOPPC event tree, where the proper success criteria and power recovery conditions are applied.

Note, the 1-FIRE-SBO fault tree is modeled only for the general transient event tree (1-FPI-OTRANS), that is, for an uncomplicated reactor trip. If it were applied to the other seven event trees (transfer trees) in Figure 9-1, the cut sets from the 1-FIRE-SBO fault tree would only propagate through the LOOP event tree and some impacts of the specific fire scenario (e.g., spurious actuations or consequential LOCA) would not be accounted for.

On the other hand, given the logic structure of the event tree shown in Figure 9-1 and the different conditions that fires can impose on the plant, the conditions where a spurious actuation or consequential LOCA (or other fire damage) occurs together with a LOOP need to be addressed. As discussed above, these conditions will not transfer through the LOOPPC event tree; therefore, the modeling does not account for the potential for offsite power recovery. To remedy this, cut set post-processing is required to account for proper offsite power recovery. Event tree post-processing rules are discussed in Section 9.4.3.2.2 and offsite power recovery assumptions are discussed in Section 9.4.3.2.4.

There is one final point to make regarding the standard fire event tree (Figure 9-1). From a review of this event tree, a single fire scenario transfers to multiple event trees developed for internal events. Therefore, when viewing results using the 'Event Tree' cut set viewing features and reports in SAPHIRE, some over-counting can occur because for individual event trees, the

[4] The "@" symbol on the event tree is a designator within SAPHIRE to not generate cut sets for that sequence.

cut sets are generated at a sequence level and not at the event tree level. Based on the individual sequence cut set generation, the overall CDF that is calculated by adding the CDF from the individual accident sequences will overestimate the CDF (i.e., there is no Boolean algebra reduction among the cut sets from separate event tree sequences). This is not typically an issue when analyzing a single event tree; however, under the current modeling process there are eight event trees analyzed concurrently. By analyzing multiple event trees together, and including fire-induced component failures, multiple redundant cut sets are generated. To rectify this over-estimation, all fire scenario cut sets are gathered into a single end state. The SAPHIRE end state gather function performs a cut set reduction (minimization) across all event tree sequences contributing to the end state. This end state (1-CD-FRI) can then be viewed to see the minimal cut sets that lead to core damage. Within the 1-CD-FRI end state, SAPHIRE retains the accident sequence that the cut sets were generated from and there are display options that will group the cut sets together based on their respective accident sequence. Placing the cut sets back into their respective sequence permits the overall sequence frequency to be calculated and facilitates the review of the final results.

9.4.3.2.2 Event Tree Post-Processing Rules

Post-processing rules are used to perform many different operations on the cut sets that result from quantifying a PRA model using SAPHIRE. Examples include the removal of disallowed maintenance combinations (e.g., diesel generator A and diesel generator B out for maintenance at the same time) and the removal of illogical cut set combinations (e.g., a cut set with LOCAs occurring simultaneously in more than one cold leg). The removal of these cut set combinations requires a strong understanding of the model, what combinations are not allowed based on technical specifications, and how the logic model generates certain combinations that need to be removed.

In many instances, a human failure event (HFE) will have a human error probability (HEP) under fire conditions that differs from that used in the internal event PRA, due to the impact the fire may have on various factors that affect human performance (e.g., timing or stress). As such, another area where post-processing rules are used is to replace Level 1 internal event PRA HFEs with their corresponding fire-related HFEs. These rules search for each Level 1 internal event operator action and then replace it with a new fire-related operator action, if appropriate.

Post-processing rules are also used to account for dependencies between multiple operator actions that are within a single cut set. The post-processing rules are designed to search for such occurrences and replace the independent HFEs with new dependent HFEs, as appropriate (i.e., the first independent HFE in the cut set would be retained as an independent action; only subsequent HFEs in the cut set would be replaced with dependent versions). The treatment of dependent operator actions is discussed in Section 16.4.

In addition, post-processing rules are used to apply different recovery actions to specific cut sets. As discussed above, there are conditions that a fire may impose on the plant that cannot readily be evaluated through the logic structure. The most straightforward way to compensate for this is to use post-processing rules. A specific example of this is the treatment of LOOP/SBO cut sets conditioned on spurious actuations and consequential LOCAs. Under these conditions, the logic structure of the L3-FPRA model does not transfer the cut sets through the LOOPPC event tree, even though a LOOP occurs, and instead forces them through one of the other transfer event trees. As such, a post-processing rule was developed to apply the operator action to align the alternate source of offsite power to these cut sets.

9.4.3.2.3 Example Target Set/Flag File Implementation

Each fire scenario event tree includes a corresponding set of "linkage rules" that can make top event substitutions, activate a flag file, and change the end state name from 1-CD to 1-CD-FRI. Linkage rules are text instructions that tell SAPHIRE what modifications are required on a sequence-by-sequence basis.

The scenario-specific fire impacts are incorporated into the PRA model using flag files. The flag files are used to eliminate credit for equipment failed by the fire. They are also used to turn on or off blocks of logic associated with spurious equipment operations, including MSOs. Flag files within SAPHIRE can only be used to change the logical operation of an event (TRUE, FALSE, IGNORE) and cannot be used to change basic event probabilities. If basic event probabilities need to be changed (e.g., for spurious operation of a valve), this new basic event is added along with a corresponding house event to "turn on" this new basic event given the fire event.

The flag files that are used for each identified fire scenario are based on the information obtained from the CAFTA/FRANX input information. The target sets identified in the fire model were used directly in the L3-FPRA model.

9.4.3.2.4 Offsite Power and EDG Recovery Assumptions

One aspect of LOOP modeling is the crediting of diesel generator repair. The L3-FPRA does not credit diesel generator repair during SBO events, consistent with the RP-FPRA and the L3PRA Level 1 PRA model for internal events (NRC, 2022a and NRC, 2022b). However, also consistent with the L3PRA Level 1 PRA model for internal events, the L3-FPRA does credit offsite power recovery for the first two hours. In addition, credit is given in the L3-FPRA for the alternate source of offsite power as a recoverable offsite power source for fire scenarios that have induced LOOPs. The L3-FPRA uses the same modeling assumptions as used in the Level 1 internal event PRA. This credit is given if the necessary equipment/circuits associated with the alternate source of offsite power feed are not damaged due to fire. Examples of necessary equipment/circuits include the standby auxiliary transformer (SAT) and the 4.16 kV AC breaker on the low voltage side of the SAT.

The alternate source of offsite power credit is provided to fire scenarios that transfer through the LOOPPC event tree and select fires that contain spurious actuations or consequential LOCAs and a LOOP. The fire scenarios that bypass the LOOPPC event tree due to the conditions imposed on the plant use post-processing rules to apply this recovery as noted in Section 9.4.3.2.2.

9.4.3.3 Fault Tree Models

The event trees described in Section 9.4.3.2 are supported by fault tree logic. The fault tree modeling for the L3-FPRA begins with the fault trees developed for the L3PRA Level 1 internal event model. Section 9.4.3.3.1 summarizes the changes made to the internal event fault trees to account for the specific effects of fires. Section 9.4.3.3.2 describes the conditional plant response fault trees that are used in the fire scenario event tree to determine which of the eight event trees in the internal event model to transfer to.

9.4.3.3.1 Summary of Revisions to Account for Fires

New logic was added to the L3PRA Level 1 internal event PRA model to reflect the impacts of fires in various locations. This new logic can be grouped into four general types:

1. logic used to generate initiating events

7. logic associated with fire-induced spurious actuations including MSO

8. instrumentation and indication, potentially impacted by fires, that is needed by the operators to perform various manual functions

9. basic events set to FALSE in the base model, which are set to TRUE during related fires to fail localized pieces of equipment during specific fire scenarios

9.4.3.3.2 Conditional Plant Response Fault Trees

Conditional plant response fault trees were developed and used in the fire scenario standard event tree (i.e., Figure 9-1). There is a fault tree that is questioned prior to each of the eight Level 1 internal event trees (e.g., see 1-FIRE-RTRIP, 1-FIRE-ISINJ, 1-FIRE-SLOCA, etc., in Figure 9-1, which are explained in detail below). These fault trees were developed to assess the impact of the fire on the plant. The fault trees either fail certain components due to the fire scenario of interest or cause spurious operation of components. For example, the fault tree 1-FIRE-SLOCA (see Figure 9-6) models a fire-induced small LOCA. One example of a fire-induced small LOCA is a spurious opening of a power-operated relief valve (PORV) and operators do not, or cannot, close the associated block valve.

Not all L3PRA Level 1 internal event PRA event trees are represented by transfer event trees in Figure 9-1. Some initiating events, such as excessive LOCA, large LOCA, and steam generator tube rupture (SGTR), cannot be feasibly hypothesized from a fire or are of such a low probability as to be ignored; therefore, no fault tree logic was developed to evaluate these events. The Level 1 internal event PRA event trees for loss of support systems (e.g., loss of a 125 volt [V] direct current [DC] bus or loss of auxiliary component cooling water [ACCW]) are also not modeled as part of the transfer event trees. The transfer to the general transient event tree captures the overall plant response for these initiators and the specific fire scenarios capture the conditional impact of losing these support systems. This eliminates the need for creating a fault tree logic model to make these adjustments and transferring to the respective event tree.

Since all fires that are modeled in the L3-FPRA will cause a reactor trip, as a minimum, they are all evaluated by transferring the fire to the reactor trip event tree (see the transfer to 1-FPI-OTRANS as sequence number 2 in Figure 9-1). Depending on the specific target set associated with each fire scenario, it may also be evaluated using one or more of the other transfer event trees in the fire scenario standard event tree. In all cases, the fire-induced failures are populated throughout the model using flag files (see Section 9.4.3.2.3).

The conditional plant response fault trees represent initiating event logic as shown in Figure 9-2 through Figure 9-10, and account for the consequences of fire due to spurious operations and fire-induced failures. Each of the conditional plant response fault trees is discussed individually below.

1-FIRE-RTRIP

The 1-FIRE-RTRIP fault tree shown in Figure 9-2, which is represented as an event tree node in Figure 9-1, is used to trigger that a reactor trip has occurred along with adjusting the mitigating system components that have failed or spuriously operated. This fault tree always triggers a reactor trip since it is assumed in the L3-FPRA that all fire scenarios will cause a reactor trip. Via the transfer gate to 1-RTRIP-F-NSBO, the logic under this fault tree queries if the fire causes spurious operation of components or fails them directly. Given the reactor trip along with any spurious operation or failure of mitigating components (that do not result in an SBO – see discussion below of 1-FIRE-SBO), the path then transfers to the Level 1 internal events model general transient event tree to generate the fire-induced core damage cut sets. The 1-IE-RTRIP-F basic event is just an event used as a designator that the fire scenario will cause a reactor trip. This basic event is turned on only when a specific fire scenario is analyzed, otherwise it is turned off (i.e., the "FALSE" indicates that it cannot occur).

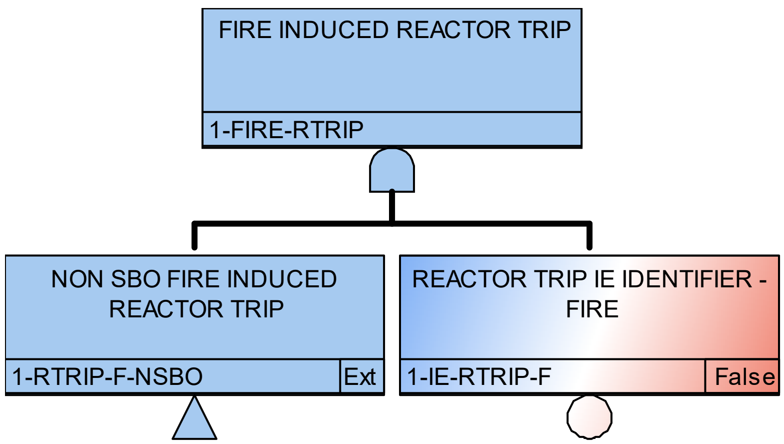

Figure 9-2 Fire-Induced Reactor Trip Logic

1-FIRE-SBO

The 1-FIRE-SBO fault tree shown in Figure 9-3 represents the last top event in Figure 9-1 and is used to ensure that when the general transient event tree path is solved, offsite power is available for the mitigating systems. This fault tree contains the logic that causes both a LOOP and failure of the onsite emergency power system (i.e., emergency diesel generators). The success path of this event tree node transfers into the general transient event tree (sequence number 2 in Figure 9-1). This ensures only non-LOOP conditions are transferred to the general transient event tree. All LOOP conditions are evaluated through the other transfers, especially the LOOPPC event tree (sequence number 10 in Figure 9-1). Forcing the LOOP conditions through the LOOPPC event tree allows for the potential recovery of offsite power.

The three transfer gates in 1-FIRE-SBO contain the logic for all possible SBO cut sets that can result following a fire. The transfer gate 1-FIRE-LOSP contains the logic representing all causes of a loss of offsite power that can result following a fire. The other two transfer gates contain the logic representing all possible failures of the onsite emergency AC (EAC) power system. The 1-EPS fault tree contains the electrical buses along with transfers to the individual diesel generators and their support systems. The 1-SBO-DGAB-F fault tree contains the diesel generator failures to start along with failures of the start relays that are related to the different fire scenarios. These relays are not captured in the Level 1 internal event modeling since the failures are spurious operations conditioned on specific fire scenarios. Therefore, both of these sub-trees need to be included in the 1-FIRE-SBO fault tree to account for both random failures of the EAC system as well as failures of the starting relays due to specific fire scenarios.

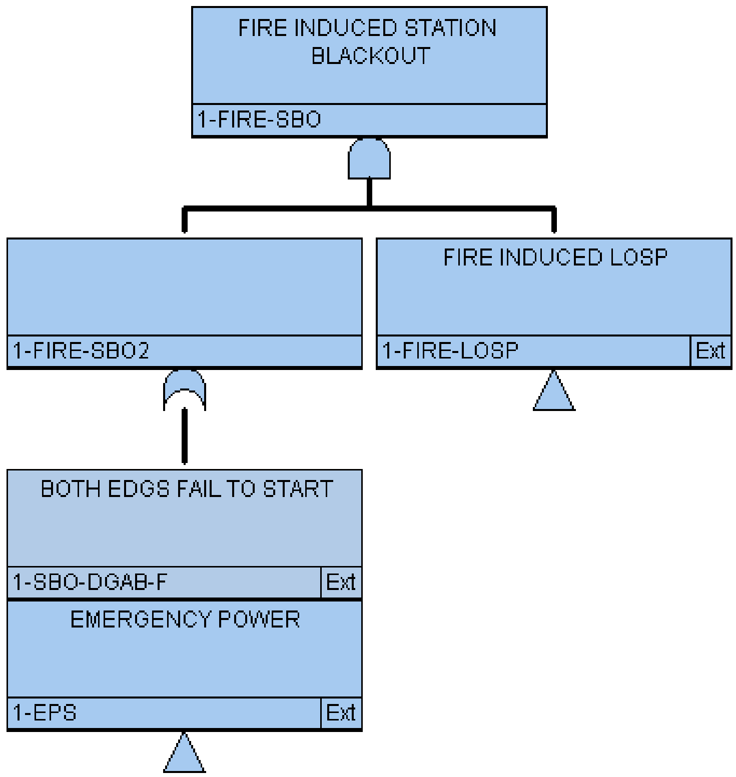

Figure 9-3 Fire-Induced Station Blackout Logic

1-FIRE-ISINJ

The 1-FIRE-ISINJ fault tree shown in Figure 9-4, which is represented as an event tree node in Figure 9-1, is used to evaluate those fire scenarios that will cause a spurious safety injection. The logic under this fault tree (transfer gate 1-ISINJ-FIRE-MSO) models the component failures that can cause an inadvertent safety injection (e.g., emergency safeguards actuation signal or charging pump operation). Given the spurious operation of any of these components (i.e., a failure of this event tree node), the sequence path transfers to the Level 1 internal events model inadvertent safety injection event tree. The 1-ISINJ-FIRE and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the inadvertent safety injection event tree or not based on the specific fire scenario. The 1-ISINJ-FIRE basic event is a designator that the fire scenario will cause a spurious safety injection signal. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

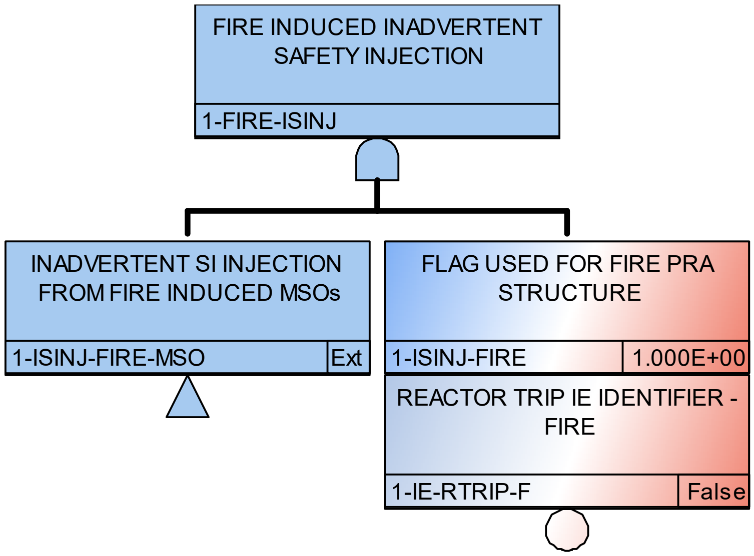

Figure 9-4 Fire-Induced Inadvertent Safety Injection Logic

1-FIRE-LOSINJ

The 1-FIRE-LOSINJ fault tree shown in Figure 9-5, which is represented as an event tree node in Figure 9-1, is used to evaluate those fire scenarios that will cause a loss of reactor coolant pump (RCP) seal injection. Given any of the fire scenarios represented by the basic events in the fault tree, RCP seal cooling is guaranteed to be failed; therefore, the logic transfers to the Level 1 internal events model loss of RCP seal injection event tree. The 1-LOSINJ-FIRE and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the loss of safety injection event tree or not based on the specific fire scenario. The 1-LOSINJ-FIRE basic event is a designator that the fire scenario will cause the loss of RCP seal cooling. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip. The logic under gates 1-FIRE-LOSINJ02 and 1-FIRE-LOSINJ03 is not shown in Figure 9-5 to reduce the size of the figure.

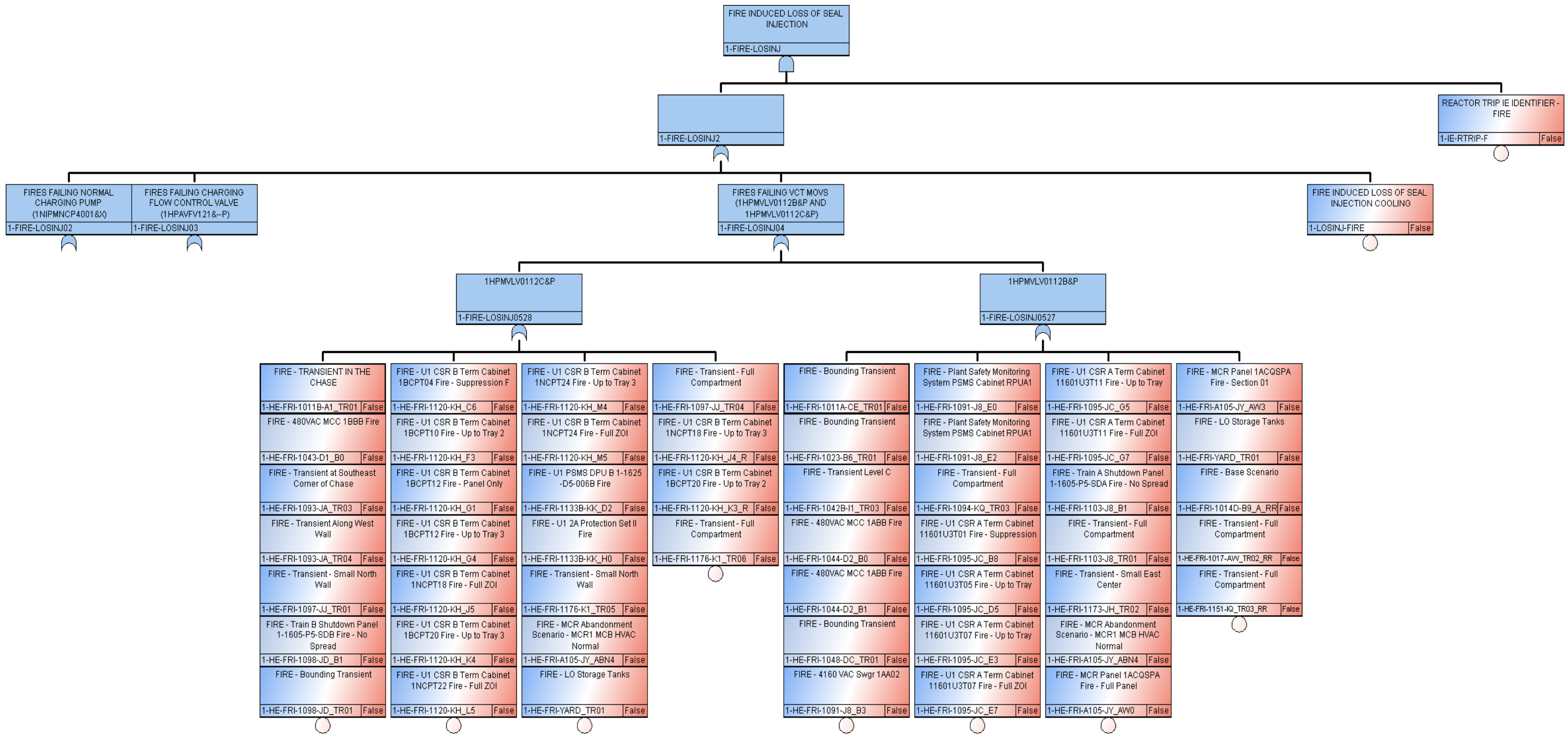

Figure 9-5 Fire-Induced Loss of Seal Injection Logic

1-FIRE-SLOCA

The 1-FIRE-SLOCA fault tree shown in Figure 9-6, which is represented as an event tree node in Figure 9-1, is used to evaluate those fire scenarios that will cause an induced small LOCA. The logic under this fault tree (transfer gate 1-SLOCA F) models those components whose failure or spurious operation can potentially cause a small LOCA. This fault tree incorporates logic associated with four distinct failure modes that can result in a small LOCA and they are:

1. spurious operation of a PORV with failure of its block valve
2. spurious isolation of letdown and relief to the pressure relief tank
3. loss of normal charging and failure to isolate letdown
4. spurious opening of reactor head vent valves

Given the fire scenario and occurrence of any of the above failure modes, a small LOCA will be induced and will be transferred to the Level 1 internal events model small LOCA event tree. The 1-SLOCA-FIRE (not shown in Figure 9-6) and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the small LOCA event tree or not based on the specific fire scenario. The 1-SLOCA-FIRE basic event (found in the transfer gate 1-SLOCA-F) designates that the fire scenario will initiate the induced small LOCA. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

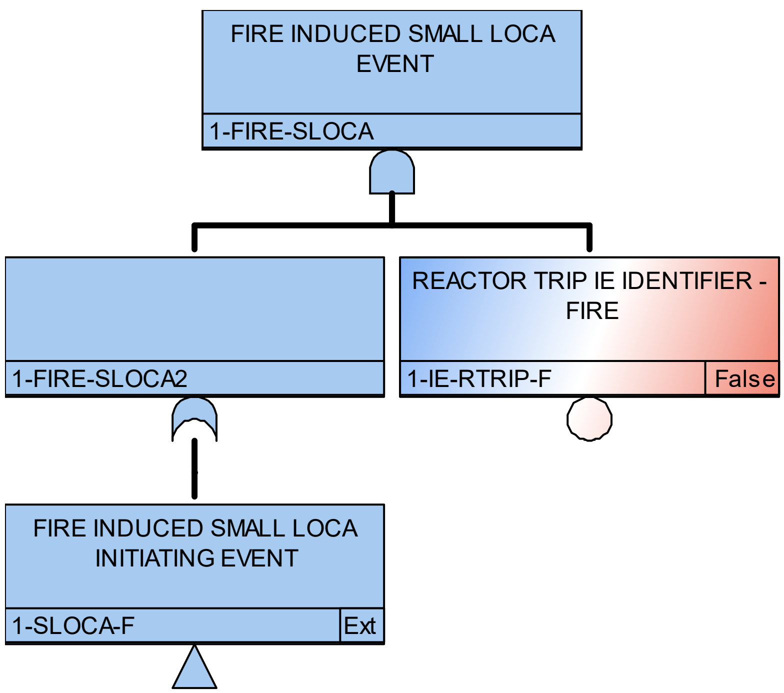

Figure 9-6 Fire-Induced Small LOCA Event Logic

1-FIRE-MLOCA

The 1-FIRE-MLOCA fault tree shown in Figure 9-7, which is represented as an event tree node in Figure 9-1, is used to evaluate the fire scenarios that will cause a fire-induced medium LOCA.

The only fire scenario identified (and modeled) that could lead to a medium LOCA is the spurious opening of both PORVs and operators do not, or cannot, close their associated block valves. This scenario is transferred to the Level 1 internal events model medium LOCA event tree. The 1-MSO-17MLOCA and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the medium LOCA event tree or not based on the specific fire scenario. The 1-MSO-17MLOCA basic event is a designator that the fire scenario will initiate the induced medium LOCA. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

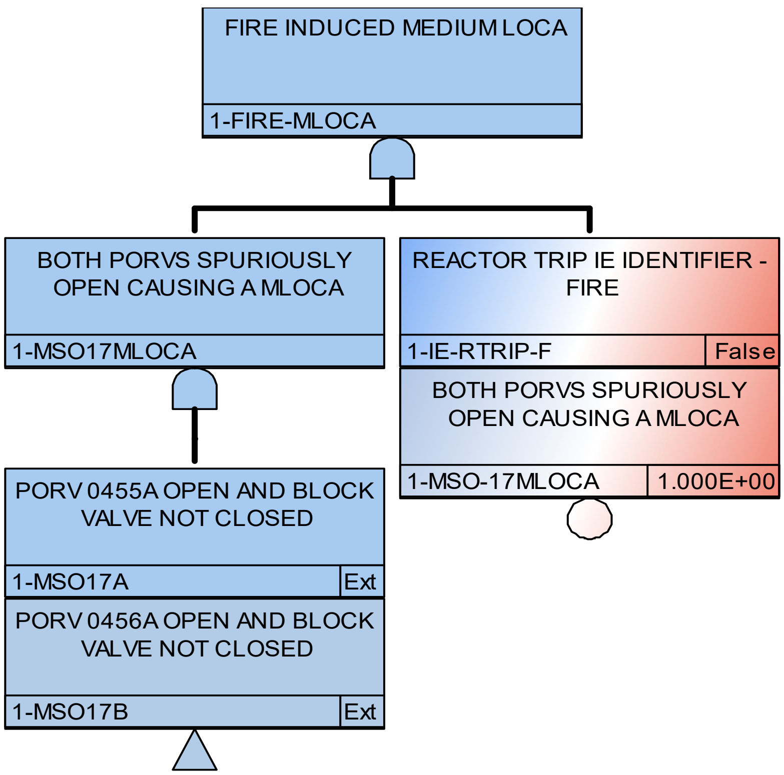

Figure 9-7 Fire-Induced Medium LOCA Event Logic

1-FIRE-CSLOCA

The 1-FIRE-CSLOCA fault tree shown in Figure 9-8, which is represented as an event tree node in Figure 9-1, is used to evaluate those fire scenarios that will cause a consequential small LOCA. This is different from the induced small LOCA discussed earlier due to the sequence (timing) of events that occur. The 1-FIRE-SLOCA and 1-FIRE-LOISNJ fault trees address LOCAs that occur as a direct consequence of the fire. The 1-FIRE-CSLOCA fault tree addresses LOCAs that occur during the plant response to a fire-induced reactor trip.

The logic under the 1-FIRE-CSLOCA fault tree models those component failures and spurious operations that will or can potentially cause a consequential small LOCA through either spurious operation of the PORVs and block valves or through loss of RCP seal cooling. Given the fire scenario and spurious operation of any of these components, a consequential small LOCA can occur and will be transferred to the Level 1 internal events model consequential small LOCA event tree. The 1-MSO-LOCAL-FIRE (not shown in Figure 9-8) and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the consequential LOCA event tree or not based on the specific fire scenario. The 1 MSO-LOCAL-FIRE basic event (found in the transfer gates 1-_PND_CONSLOCAL and 1-_PND_CONSLOCAT) is a designator that the fire scenario will initiate the consequential small LOCA.[5] The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

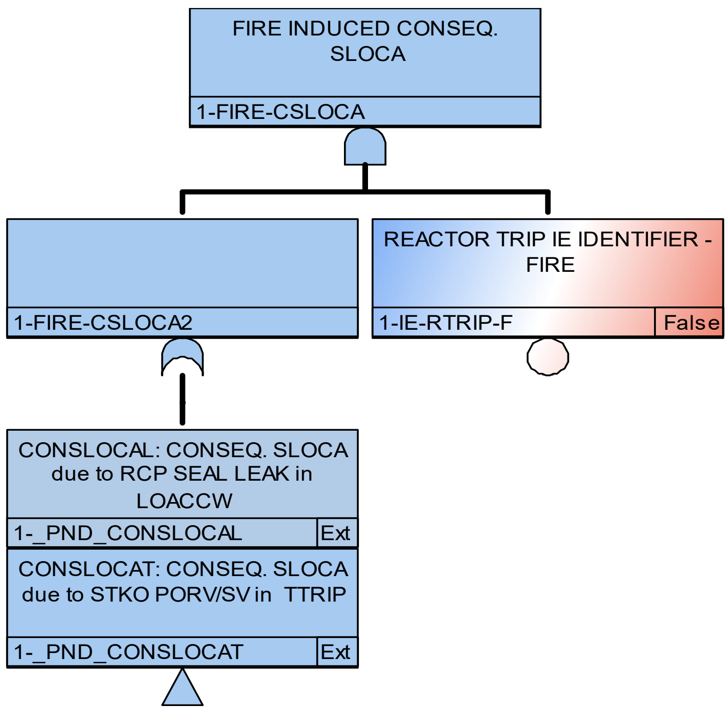

Figure 9-8 Fire-Induced Consequential LOCA Event Logic

1-FIRE-SSBI

The 1-FIRE-SSBI fault tree shown in Figure 9-9, which is represented as an event tree node in Figure 9-1, is used to evaluate those fire scenarios that will cause a steam line break upstream of the main steam isolation valves (MSIVs). The logic under this fault tree models those component failures and spurious operations that will or can potentially cause a steam line break (i.e., spurious operation of atmospheric relief valves [ARVs] or spurious operation of the auxiliary feedwater [AFW] system). Given the fire scenario and spurious operation of the components, a steam line break (or cooldown due to ARV opening) can be induced and will be transferred to the Level 1 internal events model steam line break event tree. The 1-SSBI-FIRE (not shown in Figure 9-9) and 1-IE-RTRIP-F basic events are logic switches used to either transfer to the steam line break upstream of the MSIVs event tree or not based on the specific fire scenario. The 1 SSBI-FIRE basic event (found in the transfer gate 1-_PND_SSBI-F) is a

[5] The 1-MSO-LOCAL-FIRE basic event applies to more initiating events and fire scenarios than just those referred to in the description for the 1-_PND_CONSLOCAL and 1-_PND_CONSLOCAT transfer gates. The complete logic for these transfer gates is not included here due to space considerations.

designator that the fire scenario will initiate the steam line break. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

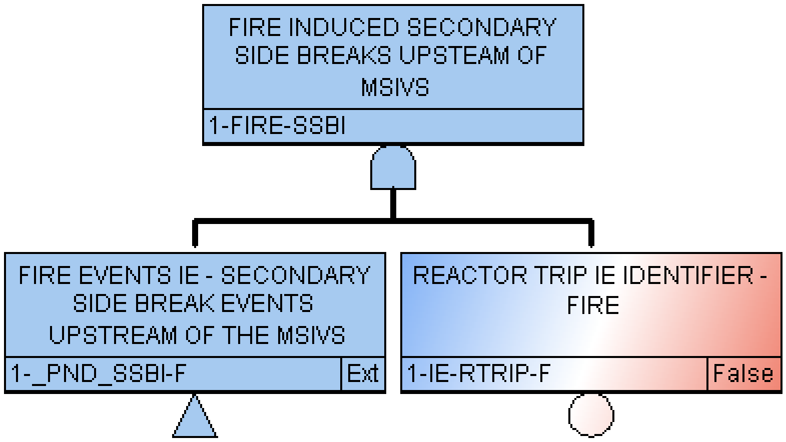

Figure 9-9 Fire-Induced Secondary Side Breaks Upstream of MSIVs Event Logic

1-FIRE-LOSP

The 1-FIRE-LOSP fault tree shown in Figure 9-10, which is represented as an event tree node in Figure 9-1, is used to evaluate LOOP events induced by a fire or occurring during plant response following a fire. Offsite power comes through the “A” and “B” reserve auxiliary transformers (RATs). Specific fire scenarios can cause a complete or partial loss of offsite power by failing one or both of the RATs (or their supply breakers) directly. A random LOOP event can occur during the plant response mission time following a fire either due to random failure of both RATs (or spurious opening of their supply breakers) or through random failures upstream of the RAT supply breakers (as represented by basic event 1-OEP-VCF-LP-RLOOP). A consequential LOOP can also occur due to transient instability on the electrical system following a reactor trip (as represented by basic event 1-OEP-VCF-LP-CLOPT). Fire-induced LOOP events and random and consequential LOOP events following a fire are transferred to the Level 1 internal events model plant-centered LOOP event tree. This event tree is specifically designed to address a plant-centered LOOP event with the appropriate recovery events. The 1-IE-RTRIP-F basic event is added to represent that all fire scenarios will cause a reactor trip.

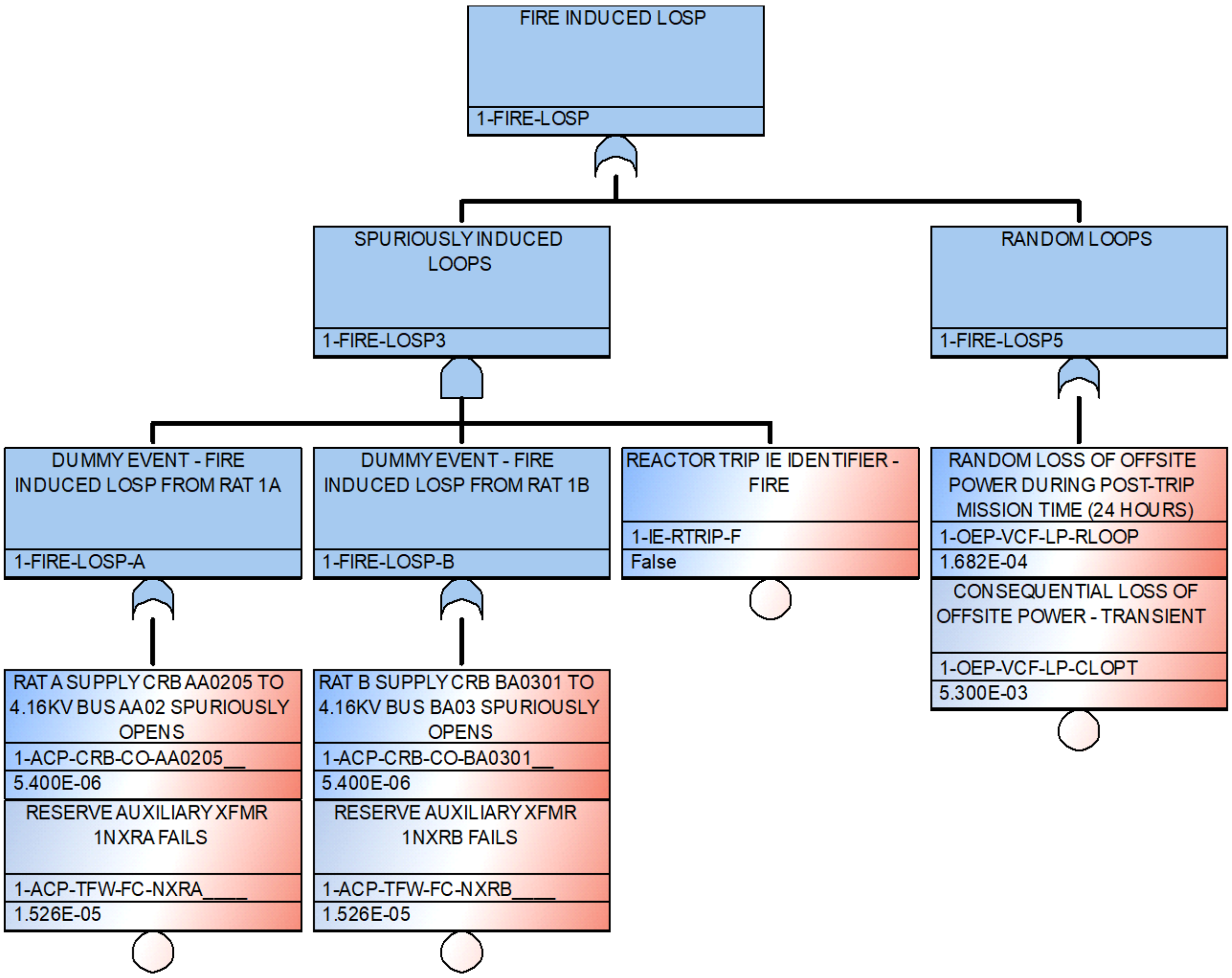

Figure 9-10 Fire-Induced LOSP Event Logic

9.4.3.4 Spurious Equipment Actuations and Valve Transfers

Spurious equipment actuations and/or valve transfers are possible during a fire due to hot shorts in control/power cabling. These uncontrolled actuations can have deleterious impacts on plant operation. Postulated spurious actions include:

- spurious pressurizer PORV opening leading to a SLOCA
- spurious emergency safeguards actuation signal
- spurious operation of breakers (potentially leading to a LOOP)
- spurious repositioning of valves
- spurious starting of AFW pumps
- combinations of the above (i.e., multiple spurious operations)

The L3-FPRA model relies heavily on the work already performed by the reference plant with respect to circuit analysis and spurious actuation modeling. The circuit analysis and spurious actuation methodology is detailed in Section 10 of NUREG/CR-6850 (NRC, 2005). In accordance with Section 10 of NUREG/CR-6850, specific components at the reference plant were identified for a detailed “Circuit Failure Mode and Likelihood Analysis.” The reference plant calculated new values for fire-induced spurious operations and incorporated these into their logic model. The logic associated with these spurious actuations was added to the L3-FPRA model and is activated using flag files. Table 9-4 lists the fault trees in the L3-FPRA model constructed to account for these spurious operations and their impacts.

Table 9-4 Fault Trees Modeling Fire-Induced Spurious Operation/Actuation

Fault Tree	Description
1-_PND_MSO-LOCAL	Fire-Induced Consequential SLOCA
1-_PND_RTRIP-F	Fire-Induced Reactor Trip Initiating Event
1-_PND_SG1-MSO	Blowdown from SG 1 Fails to Isolate
1-_PND_SG2-MSO	Blowdown from SG 2 Fails to Isolate
1-_PND_SG3-MSO	Blowdown from SG 3 Fails to Isolate
1-_PND_SG4-MSO	Blowdown from SG 4 Fails to Isolate
1-_PND_SSBI-F	Fire Events IE - Secondary Side Break Events Upstream of the MSIVs
1-ACCW-RCP-RETURN	Failure of Valves in Common ACCW Return Line from RCPs
1-ACCW-RCP-SUPPLY	Failure of Valves in Common ACCW Supply Flow for RCP Cooling
1-ACW-MOV-2041	ACCW Return from RCP TB CLG Iso MOV HV-2041 Transfers Closed
1-AFR-SG4-TDP	Turb. Driven Pump Discharge Valve HV5120 Fails to Close
1-CCP-HPI-MOV-8438	CCP A&B Discharge Interconnect MOV HV8438 Fails Closed
1-CS-RWST-MSO	Failure of RWST
1-ECCS-SUMP-A-FIRE	Loss of Flow from ECCS Containment Sump A - Fire
1-ECCS-SUMP-B-FIRE	Loss of Flow from ECCS Containment Sump B - Fire
1-FIRE-CSLOCA	Fire Induced Consequential SLOCA
1-FIRE-ISINJ	Fire Induced Inadvertent Safety Injection
1-FIRE-LOSINJ	Fire Induced Loss of Seal Injection
1-FIRE-LOSP	Fire Induced LOSP
1-FIRE-MLOCA	Fire Induced Medium LOCA
1-FIRE-RTRIP	Fire Induced Reactor Trip
1-FIRE-SBO	Fire Induced Station Blackout
1-FIRE-SLOCA	Fire Induced Small LOCA Event
1-FIRE-SSBI	Fire Induced Secondary Side Breaks Upstream of MSIVs
1-FW-MFIV-ISOL-F	Auto FW MFIV Isolation Fails
1-FW-MFRV-ISOL-F	Auto FW MFRV Isolation Fails
1-FW-PMP-ISOL-F	FW Pump Isolation Signal Fails
1-HPI-MOV-8806	HPI Suction MOV-8806 from RWST Fails Closed
1-HPI-MOV-8813	HPI MOV 8813 Fails Closed
1-ISINJ-FIRE-MSO	Inadvertent SI Injection from Fire Induced MSOs
1-MSO01AIE	Loss of Individual RCP Thermal Barrier Cooling Valves
1-MSO01BIE	Loss of Individual RCP Thermal Barrier Cooling Valves
1-MSO01CIE	Loss of Individual RCP Seal Injection Valves
1-MSO04A-RCP1	RCP P6-001 Breakers Fail to Trip
1-MSO04B-RCP2	RCP P6-002 Breakers Fail to Trip
1-MSO04C-RCP3	RCP P6-003 Breakers Fail to Trip
1-MSO04D-RCP4	RCP P6-004 Breakers Fail to Trip
1-MSO04IE	Spurious Start or Failure to Trip of RCPs
1-MSO05IE	Loss of All Seal Cooling and Leakoff Valve Failure
1-MSO06-LETDN	Failure to Isolate Letdown Path
1-MSO13IE	Loss of ACCW and Un-isolated Letdown Fail Chg Pumps
1-MSO17A	PORV 0455A Open and Block Valve Not Closed

Table 9-4 Fault Trees Modeling Fire-Induced Spurious Operation/Actuation

Fault Tree	Description
1-MSO17A-PORV455A-FIRE	PZR PORV PV-0455A Spuriously Opens Due to Fire
1-MSO17B	PORV 0456A Open and Block Valve Not Closed
1-MSO17B-PORV456A-FIRE	Pressurizer PORV PV-0456A Spuriously Opens Due to Fire
1-MSO17MIE	Both PORVs Spuriously Open Causing a MLOCA
1-MSO19IE	Spurious Opening of Reactor Head Vent Valves
1-MSO26AIE	Spurious Operation of Turbine Drive AFW Pump
1-MSO26SPUR	Spurious Steam Inlet Valve Operation or ESFAS Signal
1-MSO32-AFWA	Spurious MD AFW A
1-MSO32-AFWA-MDP-4003	Spurious Operation of MD AFWP A (Fire Related)
1-MSO32-AFWB	Spurious Operation of MD AFW B
1-MSO32IE	Secondary Side Overfill Due to Spurious Operation of Condensate or FW
1-MSO35D	Failure of Pressurizer Heaters
1-MSO-TDAFW	Spurious Operation of Turbine Drive AFW Pump
1-RTRIP-F	Reactor Trip IE Identifier - Fire
1-RTRIP-F-NSBO	Non-SBO Fire-Induced Reactor Trip
1-RWST-MSO-1	Fire-Induced MSO Results in RWST Drain Down
1-RWST-MSO-1	Fire-Induced MSO Results in RWST Drain Down
1-SGI-FWP-ISOL-1-F	FW Pumps Fail to Trip
1-SGI-MFIV-ISOL-1-F	Failure to Isolate Main Feed Water to SG _PND_ 1- Fire
1-SGI-MFIV-ISOL-2-F	Failure to Isolate Main Feed Water to SG _PND_ 2- Fire
1-SGI-MFIV-ISOL-3-F	Failure to Isolate Main Feed Water to SG _PND_ 3- Fire
1-SGI-MFIV-ISOL-4-F	Failure to Isolate Main Feed Water to SG _PND_ 4- Fire
1-SGI-MSIV-ISOL-107	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-117	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-207	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-217	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-307	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-317	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-407	MSIV or Bypass Valve Spurious Open
1-SGI-MSIV-ISOL-417	MSIV or Bypass Valve Spurious Open
1-SLOCA-F	Fire-Induced Small LOCA Initiating Event
1-SSBI-1-MSO22	Spurious Opening of ARV PV3000
1-SSBI-2-MSO22	Spurious Opening of ARV PV3010
1-SSBI-3-MSO22	Spurious Opening of ARV PV3020
1-SSBI-4-MSO22	Spurious Opening of ARV PV3030
1-VCT112B	VCT Isolation MOV LV0112B Closes

9.4.3.5 Other Data

Numerous basic events were added to the L3PRA project’s Level 1 internal event PRA model in preparation for evaluating the impact of fires. These basic events were modeled to represent cabling and components that may not have been originally modeled in the internal event PRA model. The added components and cables affect the mitigating systems and need to be accounted for when addressing fire events.

The basic events associated with the additional cable and component failures are set to FALSE (failure probability of 0.0) in the base model and inserted under OR gates, thus not changing the results of the Level 1 internal events base PRA model. Under specific fire conditions, these events are set to TRUE (failure probability of 1.0) when the associated cables or components are potentially damaged. These fault tree logic changes have no impact on the Level 1 internal events base PRA model.

10 TASK 6 – FIRE IGNITION FREQUENCIES

10.1 Objective of the Task

The objective of Task 6 is to estimate the fire-ignition frequencies. This task outlines the development of the fire-ignition frequencies starting with generic information and then updating it using plant-specific information.

10.2 Reference Plant Work Performed on the Task

The reference plant performed the analysis to determine the fire ignition frequencies for use in the RP-FPRA. the RP-FPRA documentation states that the key steps in the development of fire ignition frequencies include:

- a plant walkdown to confirm partitioning and identify fire sources
- transient fire ignition frequency development based on engineering judgment from site personnel (or "expert panel") who are familiar with the daily activities of the plant
- Bayesian updating of generic frequencies based on most current site-specific data

The RP-FPRA documentation provides a review of the fixed and transient ignition sources for each fire compartment, which were used to determine the RP-FPRA fire ignition frequencies. In addition, the RP-FPRA documentation discusses how the generic fire ignition frequencies from EPRI TR 1016735 (EPRI, 2008) were updated using plant-specific information, employing the methodology from NUREG/CR-6850, and provides the updated frequencies.

10.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL performed a review of the ignition frequency task and concluded that the fire ignition frequencies that were determined by the reference plant would be suitable for the L3PRA project. They did note, however, that the reference plant used a 0.01 severity factor for hot-work-caused fires that was applied for human failure events that result in igniting combustible material and failure of the fire watch. The review discusses the potential for under-estimating these fire ignition frequencies, since credit for both of these factors is already included in the calculation of the base fire frequency. However, the review notes that these fires would still not be expected to be risk significant contributors and the issue is noted mainly for completeness.

10.4 L3-FPRA Approach to Address the Task

A separate fire ignition frequency analysis was not performed for the L3-FPRA. Instead, the L3PRA project started with the RP-FPRA initiating event frequencies that were provided by the reference plant in the FRANX output file. These fire sequence initiating event frequencies were obtained by multiplying the fire ignition frequency by the severity factor and the non-suppression probability, as discussed in Section 15.4.1. NUREG/CR-6850 (NRC, 2005) defines severity factor as "the fraction of the fire intensity distribution that lies above the minimum fire intensity leading to fire spread and damage. The severity factor is calculated for each unique fire scenario based on the specific conditions relevant to that scenario (e.g., proximity of secondary combustibles, proximity of damage targets, damage target failure criteria, compartment conditions, etc.)." This is essentially the conditional probability that the fire will result in target damage. The non-suppression probability is the probability that neither automatic nor manual suppression system successfully terminates the fire before target damage. By using the RP-FPRA fire sequence initiating event frequencies, the RP-FPRA fire ignition frequencies were incorporated by default.

11 TASK 7 – QUANTITATIVE SCREENING

11.1 Objective of the Task

The objective of Task 7 is to perform a quantitative screening analysis. The fire compartments that were identified during Task 5 can be screened out using the quantitative screening methods discussed in NUREG/CR-6850 (NRC, 2005).

11.2 Reference Plant Work Performed on the Task

The reference plant reviewed all the identified fire compartments and did not screen any of these out based on the criteria documented in NUREG/CR-6850 (NRC, 2005). From review of the RP-FPRA documentation, there was no single report provided that discusses how this task was performed. However, the RP-FPRA peer review implies that no quantitative screening was performed and a subsequent RP-FPRA focused peer review for the qualitative and quantitative screening tasks verified that no quantitative screening was performed.

11.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the quantitative screening process and stated there would be no impact on the L3PRA given that the RP-FPRA retained all PAUs for analysis. SNL did recommend that truncating the sequence set carried from the RP-FPRA into the L3-FPRA could help limit the number of sequences needing to be analyzed in the L3-FPRA without losing important contributors.

11.4 L3-FPRA Approach to Address the Task

A separate quantitative screening analysis was not performed for the L3-FPRA, which relied on the RP-FPRA information. All the identified fire compartments (PAUs) in the RP-FPRA were carried into the L3-FPRA model for evaluation, as discussed in Sections 9 and 15.

12 TASK 8 – SCOPING FIRE MODEL

12.1 Objective of the Task

The objective of Task 8 is to develop the fire PRA framework via scoping fire modeling. The scoping fire modeling uses tools to help identify ignition sources that may impact fire risk. This task has two main objectives: (1) screen out fixed ignition sources that do not pose a threat to targets within a fire compartment and (2) assign severity factors to unscreened fixed ignition sources.

12.2 Reference Plant Work Performed on the Task

The reference plant performed this task and, due the interactions between NUREG/CR-6850 Task 8 (Scoping Fire Modeling) and Task 11 (Detailed Fire Modeling) in defining the set of fire scenarios for detailed modeling, these two tasks are discussed together in the RP-FPRA documentation. The RP-FPRA documentation discusses the general foundation that is used to perform the scoping analysis. In particular, the documentation identifies the following assumptions and uncertainties related to fire scenario selection (corresponding high-level or supporting requirements from the ASME/ANS PRA Standard [ASME, 2009] are provided in parentheses):

1. Fire scenario selection was based on plant walkdowns and cable routing data from the reference plant cable database. Target sets were defined visually based on the 98th percentile fire zone of influence based on the upper bound NUREG/CR-6850 HRRs as documented in reports that support the RP-FPRA documentation. The visual identification of targets and use of the zone of influence introduce uncertainty of the risk impact of individual scenarios. While there is potential to omit a target based on visual identification, the practice is intended to be conservative and uses a conservative zone of influence to provide reasonable assurance that potential fire impacts are evaluated. (HLR FSS-A)

2. Control Room panel impacts were assessed based on the functions at the panel and supplemented by circuit analysis. Panel fires at the Control Board are treated consistent with the in the guidance of NUREG/CR-6850, Appendix L. There is uncertainty given the results provided in NUREG/CR-6850, Appendix L are based on a certain Control Board configuration. (HLR FSS-A6)

3. Each panel in the Control Room is analyzed for the potential of a fire to exceed temperature and optical density thresholds based on the NUREG/CR-6850 criteria. As discussed in a report that supports the RP-FPRA documentation, there is uncertainty associated with the thresholds, as well as the uncertainties in the calculation identified in the report. (HLR FSS-B1)

4. Control Room abandonment risk is assessed using a bounding CCDP of 1.0 due to the low abandonment probabilities. The use of the 1.0 CCDP is conservative and should be further evaluated if abandonment scenarios become significant. (HLR FSS-B)

5. In general, the fire impacts of an ignition source are bounded by the 98th percentile fire based on NUREG/CR-6850 upper bound HRRs and assuming peak HRR at t=0. The

ignition source fire impacts are refined to a multipoint treatment when required. (SR FSS-C1, SR FSS-C3)

6. Fire modeling is based on the results of a report that supports the RP-FPRA documentation, which uses NUREG/CR-6850 HRRs. The report identifies uncertainties in the fire model used. (SR FSS-C3, HLR FSS-D)

7. Bounding fire modeling based on the results of the report that supports the RP-FPRA documentation is generally applied. The fire modeling is refined to consider point estimate HRRs and fire growth and manual suppression when required. Electric panel fire growth is based on NUREG/CR-6850. The applied HRRs and fire growth are input parameters with uncertainty that could result in conservative or non-conservative fire risk. Manual non-suppression probabilities are based on NEI-04-02 FAQ 08-0050 (NRC, 2009b). Manual detection is assumed at t=0 when automatic detection is available and not credited otherwise. (SR FSS-C1, SR FSS-C2, SR FSS-C3)

8. The reference plant has IEEE-383 qualified cables. Cable types were reviewed to determine that at least less than 5% of the cables are thermoplastic. Damage criteria for thermoset cables are assumed. (SR FSS-C5)

9. A lower transient HRR is justified for plant areas with limited open space and equipment where safety related equipment and cables are located. (SR FSS-D6)

10. Automatic detection and suppression systems are credited based on the fire hazard analysis (FHA) (SR FSS D7)

11. Screening criteria were used for the multi-compartment fire analysis (MCA) based on the guidance in NUREG/CR-6850. A screening CDF of 10% of the total CDF was used assuming that refined analysis would reduce the CDF such that the scenario is negligible. (SR FSS-G2)

12. Fire barriers were credited in the MCA consistent with the FHA. (SR FSS-G4)

13. The Generic Fire Modeling Treatments report provides sensitivity studies that show bounding fire modeling parameters were used. The RP-FPRA applied bounding parameters for other scenario specific considerations consistent with NUREG/CR-6850 and industry guidance. As such, the use of bounding parameters precludes the need for parametric uncertainty for fire modeling inputs. If certain parameters are identified as not bounding, then parametric uncertainty for those parameters should be considered consistent with industry uncertainty guidance.

12.3 SNL Review of the Reference Plant’s Approach to Address the Task

SNL reviewed the information provided by the reference plant on their development of fire modeling as required by Task 8. The review notes that the RP-FPRA documentation identifies three methods that were used for the RP-FPRA that deviate from NUREG/CR-6850 and were approved by an industry expert panel. The review also notes that none are expected to have a significant impact on the L3-FPRA. However, the review also provided several other observations that it noted might have some implications for the L3-FPRA. A synopsis of these observations is provided below:

1. The RP-FPRA fire scenario analyses are largely based on the 98th percentile screening fire intensities with no growth profile assumed (peak heat release rate (HRR) at time=0).

2. The assessment of fire damage zones is based on judgment rather than on explicit fire growth and damage modeling.

3. The RP-FPRA approach does not appear to include consideration of fire spread to secondary combustibles.

4. The RP-FPRA documentation states that, “Manual detection is assumed at t=0 when automatic detection is available and not credited otherwise.” The reviewers were unable to clearly discern the meaning and impact of this statement and stated that it may represent an optimism in the analysis.

12.4 L3-FPRA Approach to Address the Task

A separate fire scoping model and detailed fire analysis were not performed for the L3-FPRA, which used the fire scoping and detailed fire analysis directly from the RP-FPRA, as discussed and developed in the RP-FPRA documentation.

Sections 9 and 15 of this report discuss the details of the L3-FPRA fire model developed for the L3PRA project.

13 TASK 9 – DETAILED CIRCUIT FAILURE ANALYSIS

13.1 Objective of the Task

Task 9 provides the method and instructions on developing a detailed circuit failure analysis.

13.2 Reference Plant Work Performed on the Task

The reference plant performed this evaluation. As discussed previously in Section 7.2 of this report, the reference plant performed Tasks 3 and 9 concurrently based on industry guidance in NUREG/CR-6850 (NRC, 2005). As stated in the RP-FPRA documentation, "[e]ach circuit analysis includes a full disposition of cables and power supplies required to support the credited function of the component. The methodology can be subdivided into the following three tasks: Circuit Analysis, Cable Routing, and Verification of Power Supply Coordination."

The RP-FPRA documentation contains a project instruction for performing the circuit analysis. This analysis includes the following steps:

- identify cables associated with fire PRA components as per Task 3 of NUREG/CR-6850
- analyze the component response to postulated conductor/cable failures and identify those cables that can affect the credited function for the component under analysis
- screen cables that do not affect the credited component functionality
- determine the route point locations (i.e., locations of cable trays, conduits, pull boxes, etc.) for those cables required to support the credited function of the component
- identify electrical power supplies required to support safe shutdown and fire PRA components
- correlate cables and respective equipment to route points
- conduct technical checking of circuit analysis and cable routing
- document analysis results within the ARCTM FDMTM Database

The RP-FPRA documentation also states that each primary circuit conductor is evaluated for the effects of a hot short, short-to-ground, open circuit, and line-to-line fault that will prevent the desired final position of the component. The disposition of the conductors also includes evaluation of the effects of proper polarity hot shorts and multiple hot shorts. Note it is not clear whether consideration was given to the ground fault equivalent hot short (GFEHS) failure mode. GFEHS failures were not addressed in NUREG/CR-6850, Supplement 1 (NRC, 2009c) – they were first introduced in NUREG/CR-7150, Vol. 2 (NRC, 2014).

The RP-FPRA documentation identifies the following two general assumptions for the analysis:

- Equipment is assumed to be in its normal expected position or condition at the onset of the fire. In cases where the status of a component is indeterminate or could change as a result of expected plant conditions, worst-case initial conditions were assumed for the purpose of cable selection.

- Properly sized and coordinated electrical protective devices are assumed to function in accordance with their design tripping characteristics, thereby preventing initiation of secondary fires through circuit faults created by the initiating fire.

13.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the reference plant's detailed circuit failure analysis and identified no concerns or issues with the circuits analyzed. However, the review identifies several circuits that were not analyzed in detail in the RP-FPRA, and that may warrant additional analysis as part of the L3-FPRA. These include circuits for certain ISLOCA paths, the containment spray system, and some other excluded systems.

13.4 L3-FPRA Approach to Address the Task

A separate detailed circuit failure analysis was not performed for the L3-FPRA, which relied on the information provided by the reference plant.

14 TASK 10 – CIRCUIT FAILURE MODE LIKELIHOOD ANALYSIS

14.1 Objective of the Task

Task 10 provides the method and instructions on conducting a circuit failure mode likelihood analysis. This task is used to estimate the likelihood (probability) of cable failure modes due to fire-induced cable damage (e.g., hot short).

14.2 Reference Plant Work Performed on the Task

The reference plant performed this evaluation, and the documentation provides the list of components selected for circuit failure mode and likelihood analysis and the circuit failure probabilities. The circuit failure probabilities are based on the most current data from NUREG/CR-7150, Volume 2 (NRC, 2014). Tables 4-1 and 4-3 of NUREG/CR-7150, Volume 2 were the source of the mean spurious operation probabilities applied in the RP-FPRA for ungrounded direct current (DC) solenoid-operated valves (SOV) and grounded alternating current (AC) motor-operated valves (MOV), respectively, and they are shown in Table 14-1.

Table 14-1 Summary of Conditional Probability of Spurious Operation

Hot Short Failure Mode	RP-FPRA Probability
DC SOV Aggregate	0.56
DC SOV Intercable Short	0.0063
AC MOV Aggregate	0.28
AC MOV Intercable Short	0.0088

14.3 SNL Review of the Reference Plant’s Approach to Address the Task

SNL reviewed the reference plant’s documentation and their process to address this task. The review stated that the reference plant used common industry modes and values and also used current reference guidance—NUREG/CR-7150, Volume 2 (NRC, 2014).

14.4 L3-FPRA Approach to Address the Task

A separate circuit failure mode likelihood analysis was not performed for the L3-FPRA, which relied on the RP-FPRA information. The conditional probabilities and information provided by the reference plant were used directly in the L3-FPRA model.

15 TASK 11 – DETAILED FIRE MODELING

15.1 Objective of the Task

The objective of Task 11 is to perform the detailed fire modeling of all fire compartments that were identified during Task 8. The detailed fire analysis evaluates fire growth, fire propagation, and fire suppression for each fire scenario to be analyzed.

15.2 Reference Plant Work Performed on the Task

The reference plant performed this detailed fire analysis, which involves three key items: (1) single compartment (PAU) fire scenarios; (2) main control room (MCR) fires; and (3) multi-compartment fire scenarios. From the analysis, the reference plant identified 443 PAUs, which correspond to 3,306 unique fire sequences that were analyzed.

The reference plant fire PRA fire scenario identification task was approached in two steps:

1. Plant walk downs were performed to identify the set of postulated fixed and general transient ignition source fires. Each fixed ignition source identified in the fixed ignition source count effort was analyzed. In addition, general transient fires were postulated throughout the PAU at points of interest (e.g., risers and low cable trays). In general, transients were not postulated at fixed ignition sources since the fire ignition frequency of the fixed ignition source bounds that of the transient.

2. For each ignition source, a set of targets was identified based on a predefined zone of influence (ZOI) for a 98th percentile fire. The target set for each PAU was obtained from the cable routing data from the plant cable database. Each target was located to determine if a postulated fire source could damage the target.

The RP-FPRA documentation provides additional information on fire ignition frequency, target identification, damage criteria, sensitive electronics, heat release rate, fire growth, fire severity, suppression and detection, as well as several other considerations.

The RP-FPRA documentation discusses control room fire scenarios. Consideration is given to both control room fire scenarios that lead to abandonment and those that do not (i.e., those that do not exceed the habitability abandonment criteria). Fires were analyzed for each panel in the control room, including control board fires and electric panel fires. The control board fires were analyzed consistent with the guidance in Appendix L of NUREG/CR-6850, and the electric panel fires were analyzed consistent with the treatment of electric panels in other plant locations.

The RP-FPRA documentation notes that the multi-compartment analysis was completed using the FHA report, and that the RP-FPRA used the same PAU and designations

Task 11 also involves the development of fire sequence initiating event frequencies. The RP-FPRA fire sequence initiating event frequencies were calculated by multiplying the fire ignition frequency by the severity factor and the non-suppression probability. The fire ignition frequency used in this calculation is discussed under Task 6 (see Section 10). The severity factor is defined in NUREG/CR-6850 as "the fraction of the fire intensity distribution that lies above the minimum fire intensity leading to fire spread and damage. The severity factor is calculated for each unique fire scenario based on the specific conditions relevant to that scenario (e.g., proximity of secondary combustibles, proximity of damage targets, damage target failure criteria, compartment conditions, etc.).” This is essentially the conditional probability that the fire will result in target damage. The non-suppression probability is the probability that neither automatic nor manual suppression systems successfully terminate the fire before target damage, and this probability is listed for each fire sequence in the RP-FPRA documentation.

15.3 SNL Review of the Reference Plant’s Approach to Address the Task

SNL reviewed the information provided by the reference plant on detailed fire scenario analysis. The review primarily focused on the MCR fires and multi-compartment fire scenarios. The review pointed out that the MCR fires are expected to include three fire types: (1) fires within the control room that cause some damage but are not severe enough to force MCR abandonment, (2) fires in the MCR that force abandonment due to either habitability or loss of function, and (3) fires in other plant locations that could lead to MCR abandonment due to the loss of a sufficient set of plant control and monitoring functions. All but the last type of MCR fire were analyzed in the RP-FPRA. SNL noted that common practice was used in the evaluation of these fire sequences. However, the SNL review noted that the RP-FPRA results may be pessimistic for these fire sequences, since fire suppression and fire severity factors were not credited (i.e., they were assigned a value of 1.0). They also noted that the MCR abandonment scenarios were not dominant contributors to CDF, but were to LERF, and these may need to be evaluated in more detail.

In addition, SNL noted some comments about the multi-compartment fire sequences. The review pointed out that none of the multi-compartment fire sequences were dominant and this is expected based on how the fire compartments are defined. They did recommend that if other aspects of risk modeling (e.g., shutdown fire risk) are developed, multi-compartment fire sequences may have a larger impact and they should be reviewed in more detail.

15.4 L3-FPRA Approach to Address the Task

This task was not performed for the L3-FPRA, which relied on the information provided by the reference plant. The information that was used from the reference plant included the CAFTA model and FRANX file. The L3-FPRA model took the information from the reference plant and mapped it into the L3PRA Level 1 at-power model for internal events. The fire PRA model development for the L3-FPRA is discussed in Section 9.

While Task 11 was not performed for the L3-FPRA, some information from the RP-FPRA for this task is addressed in this section, since it is integral to the L3-FPRA. Specifically, fire sequence/scenario initiating event frequencies are addressed in Section 15.4.1, fires necessitating MCR abandonment are addressed in Section 15.4.2, and multi-compartment fires are addressed in Section 15.4.3. Also, Section 15.4.4 briefly addresses the potential for structural collapse due to fire.

15.4.1 Fire Sequence/Scenario Initiating Event Frequencies

As discussed in Section 10.4, the L3PRA project team started with the RP-FPRA initiating event frequencies that were provided by the reference plant in the FRANX output file. These fire sequence initiating event frequencies were obtained by multiplying the fire ignition frequency by the severity factor and the non-suppression probability. All 3,306 fire sequence initiating event frequencies from the FRANX output file were placed into an Excel Workbook. The mapping process discussed in Section 9.4.1 was used to group the 3,306 individual fire sequences from the RP-FPRA into a more manageable 210 fire scenarios to be analyzed in the L3-FPRA. The initiating event frequency for each of these 210 fire scenarios was calculated using the RP-FPRA fire sequence frequencies. For those cases where a RP-FPRA fire sequence was mapped directly on a one-to-one basis to a L3-FPRA fire scenario, the L3-FPRA fire scenario uses the RP-FPRA fire sequence fire initiating event frequency. The remaining L3-FPRA fire scenarios contain multiple RP-FPRA fire sequences. The fire initiating event frequency for these L3-FPRA fire scenarios is the summation of the fire initiating event frequencies for all of the RP-FPRA fire sequences grouped in each fire scenario.

For example, L3-FPRA fire scenario 1-IE-FRI-1002-AB_B0 has six RP-FPRA fire sequences mapped into it. To obtain the initiating event frequency for this fire scenario, the six fire sequence initiating event frequencies are summed together as shown in Table 15-1. Table 15-1 lists the six RP-FPRA fire sequences, including their fire ignition frequency, severity factor, non-suppression probability, and overall fire sequence initiating event frequency, as well as the fire scenario initiating event frequency that was calculated and used for the L3 FPRA fire scenario.

Table 15-1 L3-FPRA Fire Scenario Initiating Event Frequency Determination

RP-FPRA Sequence	Description	IGF (/rcy)	Severity Factor	NSP	Sequence Initiating Event Frequency (/rcy)
1002-AB_B0	480 V AC Switchgear 1AB15 Fire - HEAF	2.15E-04	1.00E+00	1.00E+00	2.15E-04
1002-AB_B1	480 V AC Switchgear 1AB15 Fire	1.99E-04	1.00E+00	3.69E-02	7.34E-06
1002-AB_B-SE	480 V AC Switchgear 1AB15 Fire - Sensitive Electron	3.47E-06	1.00E+00	1.00E+00	3.47E-06
1002-AB_C0	1000 KVA Transformer 1AB15X Fire	1.93E-04	1.00E+00	1.00E+00	1.93E-04
1002-AB_C1	1000 KVA Transformer 1AB15X Fire	1.93E-04	1.00E+00	9.40E-03	1.81E-06
1002-AB_TR01	Transient - Full Compartment	3.65E-05	1.00E+00	1.00E-03	3.65E-08
L3-FPRA Scenario					**Scenario Initiating Event Frequency (/rcy)**
1-IE-FRI-1002-AB_B0	480 V AC Switchgear 1AB15 Fire - HEAF				4.21E-04

Notes:
/ry – per reactor year
IGF – fire sequence ignition frequency
Severity Factor – conditional probability that given a fire has occurred, it will result in target damage
NSP – non-suppression [probability of non-suppression (automatic and/or manual)]

Table 15-2 provides the description and final fire scenario initiating event frequency for each of the 210 fire scenarios included in the L3-FPRA.

Table 15-2 Fire Scenario Descriptions and Frequencies

IE Name	Description	Scenario Initiating Event Frequency (/rcy)
1-IE-FRI-1002-AB_B0	480 V AC Switchgear 1AB15 Fire - HEAF	4.21E-04
1-IE-FRI-1011A-CE_TR01	Bounding Transient	4.25E-05
1-IE-FRI-1011B-A1_TR01	TRANSIENT IN THE CHASE	4.26E-06
1-IE-FRI-1014D-B9_A_RR	Base Scenario	8.33E-04
1-IE-FRI-1016-AV_A_RR	Base Scenario	1.40E-02
1-IE-FRI-1017-AW_TR02_RR	Transient - Full Compartment	1.56E-07
1-IE-FRI-1023-B6_TR01	Bounding Transient	8.57E-05
1-IE-FRI-1025-BT_A_RR	Base Scenario	4.45E-03
1-IE-FRI-1026A-C7_A_RR	Base Scenario	6.35E-04
1-IE-FRI-1030-C7_A_RR	Base Scenario	1.08E-02
1-IE-FRI-1031-C6_A_RR	Base Scenario	2.54E-03
1-IE-FRI-1039C-CU_TR01	Bounding Transient	4.02E-05
1-IE-FRI-1042B-I1_TR03	Transient Level C	3.22E-05
1-IE-FRI-1043-D1_B0	480 V AC MCC 1BBB Fire	5.58E-04
1-IE-FRI-1044-D2_B0	480 V AC MCC 1ABB Fire	4.46E-04
1-IE-FRI-1044-D2_B1	480 V AC MCC 1ABB Fire	1.12E-04
1-IE-FRI-1048-DC_TR01	Bounding Transient	3.78E-05
1-IE-FRI-1056A-IM_TR01RR	Transient - Full Compartment	4.44E-06
1-IE-FRI-1056B-IH_TR01RR	Transient - Full Compartment	6.11E-05
1-IE-FRI-1059-JR_A_RR	Base Scenario	4.26E-03
1-IE-FRI-1062-JM_TR09	Transient at West Wall	1.53E-06
1-IE-FRI-1066-IA_TR02	Transient - Full Compartment	1.86E-07
1-IE-FRI-1071-IF_G1_RR	480 V AC Switchgear 1BB07 Fire	1.14E-05
1-IE-FRI-1073-I7_TR03RR	TRANSIENT MIDWAY NORTH WALL	9.50E-06
1-IE-FRI-1074-ID_B1	480 V AC MCC 1NBS Fire	9.56E-05
1-IE-FRI-1074-ID_E	125 V DC MCC 1CD1M Fire	1.99E-04
1-IE-FRI-1075-I8_C01	480 V AC Switchgear 1AB05 Fire	1.40E-04
1-IE-FRI-1075-I8_D01	1000 kVA Transformer 1AB05X Fire - First Tray	5.40E-05
1-IE-FRI-1075-I8_E1	480 V AC MCC 1ABC Fire	1.12E-04
1-IE-FRI-1075-I8_F01	480 V AC Switchgear 1AB04 Fire	1.40E-04
1-IE-FRI-1076-IC_D	125 V DC Panel 1ND32 Fire	3.98E-05
1-IE-FRI-1076-IC_I	125 V DC Panel 1ND31 Fire	3.98E-05
1-IE-FRI-1077A-IJ_B1	125 V DC Switchgear 1CD1 Fire	2.16E-05
1-IE-FRI-1077A-IJ_TR01RR	Transient - Full Compartment	1.94E-06
1-IE-FRI-1078A-IL_C1	Battery Charger 1AD1CA Fire	1.34E-05
1-IE-FRI-1078A-IL_G_RR	125 V DC Panel 1AD11 Fire	9.10E-04
1-IE-FRI-1078A-IL_TR01	Transient - Full Compartment	1.11E-07
1-IE-FRI-1079A-I9_B1	125 V DC Switchgear 1BD1 Fire	2.46E-05
1-IE-FRI-1079A-I9_TR01	Transient - Full Compartment	4.63E-07
1-IE-FRI-1080-IS_B1	480 V AC MCC 1NBR Fire	9.56E-05
1-IE-FRI-1080-IS_G2	1000 kVA Transformer 1-1805-S3-B10X Fire - Full ZO	5.21E-05
1-IE-FRI-1080-IS_H2	480 V AC Switchgear 1NBL1 Fire - Target Damage	1.85E-04
1-IE-FRI-1080-IS_K2	480 V AC Switchgear 1NB09 Fire - Target Damage	3.49E-04
1-IE-FRI-1083-IG_TR01	Bounding Transient	3.65E-05
1-IE-FRI-1085-JF_TR01	Transient at Middle of Corridor	3.68E-06
1-IE-FRI-1086-KB_A_RR	Base Scenario	4.75E-04
1-IE-FRI-1091-J8_B100	4.16 kV AC Switchgear 1AA02 CUB. 00 Fire	7.77E-05
1-IE-FRI-1091-J8_B104	4.16 kV AC Switchgear 1AA02 CUB. 04 Fire	3.50E-04
1-IE-FRI-1091-J8_B200	4.16 kV AC Switchgear 1AA02 CUB. 00 Fire - No Target Damage	7.22E-05
1-IE-FRI-1091-J8_B212	4.16 kV AC Switchgear 1AA02 CUB. 12 Fire - No Target Damage	4.97E-04

Table 15-2 Fire Scenario Descriptions and Frequencies

IE Name	Description	Scenario Initiating Event Frequency (/rcy)
1-IE-FRI-1091-J8_B3	4.16 kV AC Switchgear 1AA02	4.10E-07
1-IE-FRI-1091-J8_C0	Train A Safety Features Sequencer Cabinet 1ACPSQ1	1.99E-04
1-IE-FRI-1091-J8_E0	Plant Safety Monitoring System PSMS Cabinet RPUA1	1.84E-05
1-IE-FRI-1091-J8_E2	Plant Safety Monitoring System PSMS Cabinet RPUA1	1.16E-07
1-IE-FRI-1092-J9_C100	4.16 kV AC Switchgear 1BA03 Cub. 00 Fire	1.01E-04
1-IE-FRI-1092-J9_C104	4.16 kV AC Switchgear 1BA03 Cub. 04 Fire	4.51E-05
1-IE-FRI-1092-J9_C113	4.16 kV AC Switchgear 1BA03 Cub. 13 Fire	1.13E-04
1-IE-FRI-1092-J9_C204	4.16 kV AC Switchgear 1BA03 Cub. 04 Fire - No Target Damage	9.67E-05
1-IE-FRI-1092-J9_C221	4.16 kV AC Switchgear 1BA03 Cub. 21 Fire - No Target Damage	6.45E-04
1-IE-FRI-1092-J9_C3	4.16 kV AC Switchgear 1BA03	4.26E-07
1-IE-FRI-1092-J9_D0	Cabinet 1BCPAR9 Fire	2.32E-05
1-IE-FRI-1092-J9_D1_RR	Cabinet 1BCPAR9 Fire - No Target Damage	4.88E-04
1-IE-FRI-1092-J9_E0	Train B Safety Features Sequencer Cabinet 1-1821-U	1.16E-04
1-IE-FRI-1093-JA_TR02	Transient at Train B Shutdown Panel Room Door	8.35E-07
1-IE-FRI-1093-JA_TR03	Transient at Southeast Corner of Chase	8.35E-07
1-IE-FRI-1093-JA_TR04	Transient Along West Wall	5.84E-06
1-IE-FRI-1094-KQ_B1	U1 Isolating Auxiliary Relay Cabinet 1ACPAR6 Fire	7.96E-05
1-IE-FRI-1094-KQ_C1	U1 Isolating Auxiliary Relay Cabinet 1NCPAR6 Fire	7.96E-05
1-IE-FRI-1094-KQ_H1	U1 Isolating Auxiliary Relay Cabinet 1ACPAR2 Fire	5.31E-05
1-IE-FRI-1094-KQ_J1	U1 Isolating Auxiliary Relay Cabinet 1CCPAR2 Fire	5.31E-05
1-IE-FRI-1094-KQ_TR03	Transient - Full Compartment	3.65E-08
1-IE-FRI-1095-JC_B5	U1 CSR A Term Cabinet 11601U3T01 Fire - Up to Tray	7.72E-06
1-IE-FRI-1095-JC_B8	U1 CSR A Term Cabinet 11601U3T01 Fire - Suppression	4.46E-07
1-IE-FRI-1095-JC_D5	U1 CSR A Term Cabinet 11601U3T05 Fire - Up to Tray	3.86E-06
1-IE-FRI-1095-JC_E3	U1 CSR A Term Cabinet 11601U3T07 Fire - Up to Tray	2.26E-05
1-IE-FRI-1095-JC_E7	U1 CSR A Term Cabinet 11601U3T07 Fire - Full ZOI	2.55E-06
1-IE-FRI-1095-JC_F1_RR	U1 CSR A Term Cabinet 11601U3T09 Fire - Panel Only	3.77E-04
1-IE-FRI-1095-JC_F4	U1 CSR A Term Cabinet 11601U3T09 Fire - Up to Tray	2.39E-05
1-IE-FRI-1095-JC_G1	U1 CSR A Term Cabinet 11601U3T11 Fire - Panel Only	4.23E-05
1-IE-FRI-1095-JC_G3	U1 CSR A Term Cabinet 11601U3T11 Fire - Up to Tray	1.87E-05
1-IE-FRI-1095-JC_G5	U1 CSR A Term Cabinet 11601U3T11 Fire - Up to Tray	3.86E-06
1-IE-FRI-1095-JC_G7	U1 CSR A Term Cabinet 11601U3T11 Fire - Full ZOI	1.27E-06
1-IE-FRI-1095-JC_J3	U1 CSR A Term Cabinet 11601U3T17 Fire - Up to Tray	7.95E-05
1-IE-FRI-1095-JC_K1	U1 CSR A Term Cabinet 11601U3T19 Fire - Panel Only	6.18E-05
1-IE-FRI-1095-JC_N3_RR	U1 CSR A Term Cabinet 11601U3T25 Fire - Up to Tray	1.39E-04
1-IE-FRI-1097-JJ_TR01	Transient - Small North Wall	5.06E-06
1-IE-FRI-1097-JJ_TR03	Transient - Small South Wall	5.06E-06
1-IE-FRI-1097-JJ_TR04	Transient - Full Compartment	5.33E-07
1-IE-FRI-1098-JD_B1	Train B Shutdown Panel 1-1605-P5-SDB Fire - No Spray	2.39E-04
1-IE-FRI-1098-JD_TR01	Bounding Transient	3.65E-08
1-IE-FRI-1099-J5_A_RR	Base Scenario	9.19E-04
1-IE-FRI-1103-J8_B1	Train A Shutdown Panel 1-1605-P5-SDA Fire - No Spray	2.39E-04
1-IE-FRI-1103-J8_TR01	Transient - Full Compartment	2.95E-07
1-IE-FRI-1113-JZ_TR03_RR	Transient - Full Compartment	6.35E-06
1-IE-FRI-1120-KH_C3	U1 CSR B Term Cabinet 1BCPT04 Fire - Up to Tray 2	1.40E-05
1-IE-FRI-1120-KH_C4	U1 CSR B Term Cabinet 1BCPT04 Fire - Up to Tray 3	1.06E-05
1-IE-FRI-1120-KH_C6	U1 CSR B Term Cabinet 1BCPT04 Fire - Suppression F	1.48E-06
1-IE-FRI-1120-KH_E4	U1 CSR B Term Cabinet 1BCPT08 Fire - Up to Tray 3	8.12E-06
1-IE-FRI-1120-KH_E5	U1 CSR B Term Cabinet 1BCPT08 Fire - Full ZOI	2.47E-06
1-IE-FRI-1120-KH_F3	U1 CSR B Term Cabinet 1BCPT10 Fire - Up to Tray 2	7.00E-06

Table 15-2 Fire Scenario Descriptions and Frequencies

IE Name	Description	Scenario Initiating Event Frequency (/rcy)
1-IE-FRI-1120-KH_G1	U1 CSR B Term Cabinet 1BCPT12 Fire - Panel Only	2.00E-05
1-IE-FRI-1120-KH_G4	U1 CSR B Term Cabinet 1BCPT12 Fire - Up to Tray 3	1.98E-05
1-IE-FRI-1120-KH_H1	U1 CSR B Term Cabinet 1BCPT14 Fire - Panel Only	2.00E-05
1-IE-FRI-1120-KH_H2	U1 CSR B Term Cabinet 1BCPT14 Fire - First Tray	1.98E-05
1-IE-FRI-1120-KH_J4	U1 CSR B Term Cabinet 1NCPT18 Fire - Up to Tray 3	1.97E-05
1-IE-FRI-1120-KH_J5	U1 CSR B Term Cabinet 1NCPT18 Fire - Full ZOI	2.47E-06
1-IE-FRI-1120-KH_K3	U1 CSR B Term Cabinet 1BCPT20 Fire - Up to Tray 2	2.96E-05
1-IE-FRI-1120-KH_K4	U1 CSR B Term Cabinet 1BCPT20 Fire - Up to Tray 3	2.83E-06
1-IE-FRI-1120-KH_L5	U1 CSR B Term Cabinet 1NCPT22 Fire - Full ZOI	4.94E-06
1-IE-FRI-1120-KH_M4	U1 CSR B Term Cabinet 1NCPT24 Fire - Up to Tray 3	8.12E-06
1-IE-FRI-1120-KH_M5	U1 CSR B Term Cabinet 1NCPT24 Fire - Full ZOI	4.94E-06
1-IE-FRI-1120-KH_TR04	TRANSIENT AT RISER ROW 1 - SOUTH OF CABINETS	3.58E-07
1-IE-FRI-1120-KH_TR09	TRANSIENT AT RISER ROW 5	3.58E-07
1-IE-FRI-1121-KG_B1	U1 Isolating Auxiliary Relay Cabinet 1BCPAR3 Fire	1.19E-04
1-IE-FRI-1121-KG_D1_RR	U1 Isolating Auxiliary Relay Cabinet 1NCPAR4 Fire	3.59E-04
1-IE-FRI-1121-KG_E1	U1 Isolating Auxiliary Relay Cabinet 1BCPAR7 Fire	5.97E-05
1-IE-FRI-1121-KG_TR01	Transient - Full Compartment	4.97E-07
1-IE-FRI-1133B-KK_D2	U1 PSMS DPU B 1-1625-D5-006B Fire	4.24E-08
1-IE-FRI-1133B-KK_E0	U1 PSMS RPU-B2 1-1625-D5-004 Fire	9.30E-05
1-IE-FRI-1133B-KK_H0	U1 2A Protection Set II Fire	1.98E-05
1-IE-FRI-1140A-S1_E	Elevation 185 - North	9.54E-06
1-IE-FRI-1140A-S1_I_RR	Elevation 197 - South	1.10E-03
1-IE-FRI-1140B-S1_B	Elevation 171 - North	5.43E-04
1-IE-FRI-1140B-S1_E	Elevation 185 - North	8.61E-05
1-IE-FRI-1140B-S1_H	Elevation 171 - South	1.48E-04
1-IE-FRI-1140C-S1_C_RR	Elevation 197 - North	2.78E-03
1-IE-FRI-1140C-S1_L2	Elevation 185 - South - SG4 QUADRANT	7.74E-04
1-IE-FRI-1144-T6_JB3_RR	Junction Box 1BWJB4867	8.62E-05
1-IE-FRI-1146-VF_TR01_RR	Bounding Transient	2.76E-03
1-IE-FRI-1149-DO_TR02_RR	Transient - Full Compartment	5.87E-07
1-IE-FRI-1151-IQ_TR03_RR	Transient - Full Compartment	4.18E-06
1-IE-FRI-1152-IN_TR01_RR	Transient - Below 6 Feet	1.41E-03
1-IE-FRI-1152-IN_TR02	Transient - Full Compartment	5.04E-07
1-IE-FRI-1153-IQ_TR01	Bounding Transient	5.77E-05
1-IE-FRI-1155-V1_TR01_RR	Transient - Full Compartment	7.13E-05
1-IE-FRI-1156-V2_B	AFW Train A Pump Motor Fire	9.63E-05
1-IE-FRI-1157A-V3_B1	AFW Train C Turbine Driven Pump Fire	1.96E-05
1-IE-FRI-1160A-V8_C_RR	NSCW Train A Pump 1 Motor Fire	7.78E-04
1-IE-FRI-1161-T1_B	EDG 1A FIRE/BOUNDING TRANSIENT	3.02E-03
1-IE-FRI-1162-T2_B	EDG 1B FIRE/BOUNDING TRANSIENT	3.02E-03
1-IE-FRI-1163-T3_A_RR	Base Scenario	2.76E-03
1-IE-FRI-1173-JH_TR02	Transient - Small East Center	5.06E-06
1-IE-FRI-1174-JG_TR04_RR	Transient - Full Compartment	7.16E-05
1-IE-FRI-1175-JI_TR01	Bounding Transient	3.65E-05
1-IE-FRI-1176-K1_TR04	Transient - Small South Wall	5.06E-06
1-IE-FRI-1176-K1_TR05	Transient - Small North Wall	5.06E-06
1-IE-FRI-1176-K1_TR06	Transient - Full Compartment	3.65E-08
1-IE-FRI-1176-K1_TR07	Transient near 1BE31CTYAER2	2.55E-06
1-IE-FRI-1179-KV_B1_RR	TSC Inverter A-1807-Y3-TSCI7 Fire	1.13E-04
1-IE-FRI-1188-VH_A_RR	Base Scenario	6.57E-03
1-IE-FRI-1300A-X1_A_RR	Base Scenario	7.70E-05

Table 15-2 Fire Scenario Descriptions and Frequencies

IE Name	Description	Scenario Initiating Event Frequency (/rcy)
1-IE-FRI-1503_TR01_RR	BOUNDING TRANSIENT	1.41E-04
1-IE-FRI-1506_B1	MAIN FEEDWATER PUMP FIRE - OIL FIRE	4.56E-06
1-IE-FRI-1506_JB1	Junction Box 1NQJB6012	1.00E-04
1-IE-FRI-1507_B1	MAIN FEEDWATER PUMP FIRE - OIL FIRE	4.56E-06
1-IE-FRI-1508_TR01	TRANSIENT AT CHASE	3.90E-06
1-IE-FRI-1509_Q_RR	480 V SWITCHGEAR 1NBL2 - No Target Damage	1.75E-02
1-IE-FRI-1512_B0	Switchgear 1NAB Fire - HEAF	2.97E-05
1-IE-FRI-1512_C0_RR	Switchgear 1NB03 Fire - HEAF	1.47E-04
1-IE-FRI-1512_C2	Switchgear 1NB03 Fire	1.30E-04
1-IE-FRI-1512_D1	Transformer 11805S3B03X Fire - Damage	6.25E-05
1-IE-FRI-1512_K0	13.8 kVAC Switchgear - 1NAA HEAF	2.97E-05
1-IE-FRI-1512_K2	13.8 kVAC Switchgear - 1NAA	2.52E-04
1-IE-FRI-1530_A_RR	Base Scenario	5.28E-02
1-IE-FRI-1603-KD_E_RR	125 V DC Panel A-1806-Q3-TS2/3/C Fire	9.65E-03
1-IE-FRI-1800_A_RR	Base Scenario	4.12E-02
1-IE-FRI-2050-D4_A_RR	Base Scenario	3.29E-02
1-IE-FRI-2080-M9_H1	480 V AC MCC 2NBR Fire	1.20E-03
1-IE-FRI-2085-NB_TR04_RR	TRANSIENT 2 AT NORTH WALL	7.82E-05
1-IE-FRI-2091-N4_B100	4.16 kV AC Switchgear 2AA02 Cub. 00 Fire	3.41E-04
1-IE-FRI-2095-N8_JB1	Junction Box	1.32E-03
1-IE-FRI-2098-N9_JB1_RR	Junction Box	5.96E-04
1-IE-FRI-2115-JZ_A_RR	Base Scenario	4.19E-02
1-IE-FRI-2133A-KK_A_RR	Base Scenario	1.22E-02
1-IE-FRI-2136-LP_A_RR	Base Scenario	3.35E-03
1-IE-FRI-A040-BX_A_RR	Base Scenario	5.62E-03
1-IE-FRI-A105-JY_ABN4	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	1.40E-07 (Note 1)
1-IE-FRI-A105-JY_AI	MCR Panel 11604Q5PS3 Fire	3.98E-05
1-IE-FRI-A105-JY_AR	MCR Panel 1NCQARB Fire	7.96E-05
1-IE-FRI-A105-JY_AT0	MCR Panel 1BCQSPB Fire - Full Panel	2.79E-07
1-IE-FRI-A105-JY_AT1	MCR Panel 1BCQSPB Fire - Input Section	9.96E-06
1-IE-FRI-A105-JY_AT2	MCR Panel 1BCQSPB Fire - Logic Section	9.96E-06
1-IE-FRI-A105-JY_AT3	MCR Panel 1BCQSPB Fire - Section 01	9.96E-06
1-IE-FRI-A105-JY_AV	MCR Panel 1ACQSTA Fire	3.98E-05
1-IE-FRI-A105-JY_AW0	MCR Panel 1ACQSPA Fire - Full Panel	2.79E-07
1-IE-FRI-A105-JY_AW1	MCR Panel 1ACQSPA Fire - Input Section	9.96E-06
1-IE-FRI-A105-JY_AW2	MCR Panel 1ACQSPA Fire - Logic Section	9.96E-06
1-IE-FRI-A105-JY_AW3	MCR Panel 1ACQSPA Fire - Section 01	9.96E-06
1-IE-FRI-A105-JY_AX	MCR Panel - AMSAC 11626Q5AMS Fire	3.98E-05
1-IE-FRI-A105-JY_L	MCR Panel QPCP Fire	7.96E-05
1-IE-FRI-A105-JY_M	MCR Panel QPP1 Fire	3.98E-05
1-IE-FRI-A105-JY_P2	MCB Panel QMCB A1 Fire - NSCW	4.41E-06
1-IE-FRI-A105-JY_Q1	MCB Panel QMCB A2 Fire - RHR	4.41E-06
1-IE-FRI-A105-JY_Q6	MCB Panel QMCB A2 Fire - RHR AND LETDOWN	2.09E-06
1-IE-FRI-A105-JY_R0	MCB Panel QMCB C Fire - Full Panel	2.09E-06
1-IE-FRI-A105-JY_S2	MCB Panel QMCB B1 Fire - FW PT	8.81E-06
1-IE-FRI-A105-JY_S3	MCB Panel QMCB B1 Fire - AFW	4.41E-06
1-IE-FRI-A105-JY_S5	MCB Panel QMCB B1 Fire - MAIN STEAM AND FW	3.14E-06
1-IE-FRI-A105-JY_U3A	MCB Panel QEAB 1A Fire - SECT. 3 - FIRE SPREAD	6.17E-08
1-IE-FRI-A105-JY_U5A	MCB Panel QEAB 1A Fire - SECT. 5 - FIRE SPREAD	3.08E-08
1-IE-FRI-A105-JY_U7A	MCB Panel QEAB 1A Fire - SECT. 7 - FIRE SPREAD	3.08E-08

Table 15-2 Fire Scenario Descriptions and Frequencies

IE Name	Description	Scenario Initiating Event Frequency (/rcy)
1-IE-FRI-AHVSWYD_B	Relay Panel Fire Results in Loss of NXRA	1.59E-04
1-IE-FRI-AHVSWYD_C	Relay Panel Fire Results in Loss of NXRB	1.59E-04
1-IE-FRI-AHVSWYD_E	Main Control Panel Fire Results in Loss of Both Reserve Auxiliary Transformers	5.60E-04
1-IE-FRI-AHVSWYD_TR01	Bounding Pull Box Transient Fire	7.11E-05
1-IE-FRI-ALVSWYD_TR01	Bounding Pull Box Transient Fire	1.55E-05
1-IE-FRI-TB1_A	Multi Compartment Scenario	1.92E-06
1-IE-FRI-YARD_TR01	LO Storage Tanks	7.24E-09
1-IE-FRI-YARD_TR05	Pull Box 1NE7BBKEM02	8.00E-04
1-IE-FRI-YARD_TR10	Pull Box 2NCPXRA	2.58E-05
Note 1. The severity factor for HVAC in normal mode has been reduced by the HEP of 0.1 to account for operator failure to place the HVAC in purge mode, as discussed in Section 18.4.5.5.		

15.4.2 Main Control Room Abandonment Scenarios

The control room complex is a shared fire area for Unit 1 and Unit 2, each having its own control area. Separate alternate safe shutdown capability is provided in the form of remote shutdown panels and other local control stations for each unit.

In case of control room abandonment, an abnormal operating procedure provides operator instructions for evacuating the MCR, maintaining hot standby, and attaining cold shutdown from the remote shutdown panels. This procedure is applicable with or without the availability of offsite power. This procedure addresses potential or actual component failures that may be induced by control room fire events.

For all MCR abandonment scenarios, the RP-FPRA postulates that all equipment is failed other than those components associated with the alternate shutdown capability (ASC). Given the low probability of MCR abandonment, potential for spurious operation, and considerations related to Information Notice 92-18 (loss of remote shutdown capability) (NRC, 1992), further evaluation of ASC was not performed for the RP-FPRA and a 1.0 CCDP was applied.

Table 15-3 summarizes the twelve MCR abandonment fire scenarios modeled in the RP-FPRA, which yield a combined CDF of 8.3×10^{-7}/rcy.[6] These twelve MCR abandonment fire scenarios are modeled as one fire scenario in the L3-FPRA, since it is assumed that these fire sequences all lead to core damage with a CCDP of 1.0.

A sensitivity analysis allowing credit for the remote shutdown panel is provided in Section 19.4.3 of this report.

[6] As discussed in the RP-FPRA documentation, for MCR abandonment scenarios with the heating, ventilation, and air-conditioning (HVAC) in normal mode, successfully switching the HVAC to purge mode precludes the need to abandon the MCR. The reference plant assigned a screening human error probability of 0.1 for this action. However, as discussed in Section 18.4.5.5 of this report, the reference plant inadvertently failed to credit this action when performing their fire PRA quantification, so it is not reflected in the CDF values reported in Table 15-3. This action ***is*** credited in the L3-FPRA quantification of MCR abandonment scenarios.

Table 15-3 Main Control Room Abandonment Scenarios

Fire Area (PAU)	Scenario	Ignition Source	IGF (/rcy)	Severity Factor	NSP[1]	Sequence Frequency [IE] (/rcy) (RP-FPRA)	CDF (/rcy) (RP-FPRA)
A105-JY	A105-JY_ABN2	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	1.24E-4	1.82E-4	1.0	2.26E-8	2.26E-8
A105-JY	A105-JY_ABN3	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	2.48E-3	2.60E-5	1.0	6.45E-8	6.45E-8
A105-JY	A105-JY_ABN4	MCR Abandonment Scenario - MCR1 main control board (MCB) HVAC Normal	8.16E-4	1.60E-4	1.0	1.31E-7	1.31E-7
A105-JY	A105-JY_ABN5	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	1.24E-4	7.48E-5	1.0	9.28E-9	9.28E-9
A105-JY	A105-JY_ABN6	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	2.19E-3	2.60E-5	1.0	5.69E-8	5.69E-8
A105-JY	A105-JY_ABN7	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	8.16E-4	1.60E-4	1.0	1.31E-7	1.31E-7
A105-NO	A105-NO_ABN2	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	1.24E-4	1.82E-4	1.0	2.26E-8	2.26E-8
A105-NO	A105-NO_ABN3	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	2.19E-3	2.60E-5	1.0	5.69E-8	5.69E-8
A105-NO	A105-NO_ABN4	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	8.16E-4	1.60E-4	1.0	1.31E-7	1.31E-7
A105-NO	A105-NO_ABN5	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	1.24E-4	7.48E-5	1.0	9.28E-9	9.28E-9
A105-NO	A105-NO_ABN6	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	2.48E-3	2.60E-5	1.0	6.45E-8	6.45E-8
A105-NO	A105-NO_ABN7	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	8.16E-4	1.60E-4	1.0	1.31E-7	1.31E-7

Table 15-3 Main Control Room Abandonment Scenarios

Fire Area (PAU)	Scenario	Ignition Source	IGF (/rcy)	Severity Factor	NSP[1]	Sequence Frequency [IE] (/rcy) (RP-FPRA)	CDF (/rcy) (RP-FPRA)
/rcy – per reactor critical year IGF – fire sequence ignition frequency Severity Factor – conditional probability that given a fire has occurred, it will result in target damage NSP – non-suppression [probability of non-suppression (automatic and/or manual)] Sequence Frequency – the sequence frequency is the initiating event frequency (IGF * Severity Factor * non-suppression) CDF – core damage frequency, which is obtained through quantification of the fire sequence using the sequence frequency, fire damage vector, and plant response (PRA) model (since the conditional core damage probability for all these sequences was assumed to be 1.0, the CDF is the same as the sequence frequency) Note 1: NSP is reported as 1.0 for all scenarios because the probability of non-suppression is already accounted for in the calculation of the severity factor.							

15.4.3 Multi Compartment (Fire) Analysis

As discussed earlier, the reference plant performed an extensive multi-compartment fire analysis (MCA) as part of the RP-FPRA. Those areas that were not screened out by this analysis were modeled in the RP-FPRA for CDF quantification. Ten MCA sequences survived the screening process and were evaluated in the FRANX data base. These 10 sequences are shown in Table 15-4.

The 10 MCA fire sequences in Table 15-4 were analyzed in the RP-FPRA by addressing the target sets of the fire sequence along with the overall fire sequence initiating event frequency (ignition frequency * severity factor * non-suppression probability). From this analysis, eight of the MCA fire sequences fell below the truncation limit of 10^{-11}/rcy. As such, these sequences are listed as having a CCDP and CDF of 0.0 in the RP-FPRA quantification report and were not modeled in the L3-FPRA. The remaining two sequences, which were not truncated when evaluated in the RP-FPRA, were mapped into fire scenarios to be analyzed in the L3-FPRA.

The CDF results for the MCA scenarios are discussed in Section 18.4.5.6 of this report.

Table 15-4 MCA Sequences in RP-FPRA for Unit 1 CDF

MCA Fire Sequences Modeled in L3-FPRA (results listed are from the RP-FPRA)								
Scenario	**Description**	**Fire Area (PAU)**	**IGF (/rcy)**	**Severity Factor**	**NSP**	**Sequence Frequency (/rcy)**	**CCDP**	**CDF (/rcy)**
TB1_A	Multi Compartment Scenario	TB1	3.84E-3	5.0E-4	1.0	1.92E-6	4.89E-2	9.38E-8
TB2_A	Multi Compartment Scenario	TB2	3.84E-3	5.0E-4	1.0	1.92E-6	6.89E-4	1.32E-9
				Partial Sum =		3.84E-6	2.48E-2	9.51E-8

Table 15-4 MCA Sequences in RP-FPRA for Unit 1 CDF

MCA Fire Sequences not Modeled in L3-FPRA (results listed are from the RP-FPRA)								
Scenario	**Description**	**Fire Area (PAU)**	**IGF (/rcy)**	**Severity Factor**	**NSP**	**Sequence Frequency (/rcy)**	**CCDP**	**CDF (/rcy)**
1-6-3_1A	MCA FROM 1006-AD TO 1003-AA	1-6-3	9.63E-5	1.20E-1	1.06E-1	1.22E-6	0	0
1-6-3_TR1	MCA FROM 1006-AD TO 1003-AA	1-6-3	4.66E-5	1.20E-1	5.56E-2	3.11E-7	0	0
1-193-194_TR1	MCA FROM 1193-VM TO 1194-VL	1-193-194	2.23E-4	1.00E-1	1.88E-2	4.19E-7	0	0
1-603-604_1A	MCA FROM 1603-KD TO 1604-KD	1-603-604	3.13E-4	1.00E-1	5.20E-2	1.63E-6	0	0
2-6-3_2A	MCA FROM 2006-ED TO 2003-EA	2-6-3	9.63E-5	1.20E-1	1.15E-1	1.33E-6	0	0
2-6-3_TR2	MCA FROM 2006-ED TO 2003-EA	2-6-3	4.66E-5	1.20E-1	6.42E-2	3.59E-7	0	0
2-188-146_2A	MCA FROM 2188-W7 TO 2146-W6	2-188-146	7.96E-5	5.00E-3	3.26E-1	1.30E-7	0	0
2-188-146_2B	MCA FROM 2188-W7 TO 2146-W6	2-188-146	4.81E-4	5.00E-3	1.83E-1	4.40E-7	0	0
				Partial Sum =		5.84E-6		0
					Total =	9.68E-6		9.51E-8

Notes:
/rcy – per reactor critical year
IGF – fire sequence ignition frequency
Severity Factor – conditional probability that given a fire has occurred, it will result in target damage
NSP – non-suppression [probability of non-suppression (automatic and/or manual)]
Sequence Frequency – the sequence frequency is the initiating event frequency (IGF * Severity Factor * non-suppression)
CCDP – conditional core damage probability, which is obtained by dividing the sequence CDF by the sequence frequency [initiating event frequency]
CDF – core damage frequency, which is obtained through quantification of the fire sequence using the sequence frequency, fire damage vector, and plant response (PRA) model

15.4.4 Potential Structural Collapse

The RP-FPRA documentation includes a discussion of structural fire resistance and identifies the construction of the plant and fire-resistance of the interior, exterior, and supporting walls. This information is from the reference plant's Final Safety Analysis Report and Design Criteria report. This evaluation was provided to be consistent with the ASME/ANS PRA Standard (ASME, 2009). The RP-FPRA documentation identifies 29 different fire compartments (PAUs), excluding the Turbine Building, with the potential of containing sufficient hazard sources (i.e., a potential heat load of at least 7.0×10^6 BTU, which can lead to fire temperatures greater than 1000 ºF) to cause structural damage to exposed steel. However, 28 of the PAUs were screened out because they are Category 1 structures (e.g., they are coated with inorganic material that has been tested and approved by Underwriters Laboratories). The last PAU is part of the Radwaste Building. This building does not contain any fire PRA targets nor is it close to any buildings that contain fire PRA targets; therefore, the treatment of potential structural failures in this area are bounded by the base fire sequence.

The Turbine Building is a Category 2 building and has sufficient hazard sources to cause potential structural damage to exposed steel. Therefore, a fire sequence is modeled in the RP-FPRA that could impact the structural steel leading to its potential failure. The development of the fire sequence uses the guidance from Appendix O of NUREG/CR-6850.

Based on review of the information provided, there is no means of identifying which fire sequence is specifically modeled as having the potential for structural failure. The Turbine Building PAUs are labeled as 1500 – 1515. All of the fire sequences associated with these PAUs that are above the 10^{-12}/rcy truncation were mapped into a L3-FPRA fire scenario; therefore, the potential failure of structural steel in a Turbine Building fire is captured in one of the 210 L3-FPRA fire scenarios.

16 TASK 12 – POST-FIRE HUMAN RELIABILITY ANALYSIS

16.1 Objective of the Task

The objective of Task 12 is to identify the human failure events (HFEs) to include in the fire PRA and quantify their associated human error probabilities (HEPs). This includes both potential modification of HEPs from the Level 1 internal event PRA model, as well as inclusion of any new HFEs that are only relevant to fire scenarios. This task is conducted in two phases – a screening human reliability analysis (HRA), followed by a more detailed HRA of the most risk-significant HFEs. Note, the focus of this task is on post-fire HFEs; detailed modeling of pre-fire HFEs is not currently within the state of practice.

16.2 Reference Plant Work Performed on the Task

The reference plant performed a detailed post-fire HRA. Rather than scrub the full power internal events (FPIE) model for HFEs applicable for FPRA, all of the FPIE operator actions were retained for applicability in the FPRA; however, they were redefined in the context of a fire scenario. The RP-FPRA documentation states that a process was employed to identify additional HFEs that would only be required in response to a fire, as directed by the fire response procedures. This process involved a combination of procedure review and fire scenario/sequence development. The set of fire HFEs identified as part of this process were first assessed with conservative screening values and those that were determined to be risk significant were reevaluated using detailed HRA.

The RP-FPRA documentation also discusses the potential for undesired operator actions in response to spurious indications. However, based on guidance in NUREG/CR-6850 (NRC, 2005) and the ASME/ANS PRA Standard (ASME, 2009), and discussions with a recently retired representative from the Operations and Operations Training departments, the reference plant assumed that no such actions would be taken that would result in the change of state of equipment.

The RP-FPRA documentation describes the HRA quantification approach, which involves both an initial feasibility assessment and subsequent detailed quantifications of (1) actions that already have detailed assessments in the FPIE HRA or (2) actions with screening values that became risk significant in the fire PRA. Feasibility considerations include the availability of operator cues, procedure direction, personnel resources, and time for diagnosis and execution. Actions that fail the feasibility assessment are assigned an HEP of 1.0.

The FPIE operator actions that were evaluated using the EPRI HRA Calculator were re-evaluated for the fire PRA, generally following the guidance in Appendix C of NUREG-1921 (NRC, 2012). The RP-FPRA documentation lists the various fire impacts that were considered in the re-evaluations, as well as a set of additional considerations and assumptions.

Lastly, the RP-FPRA documentation describes the HRA dependency analysis that was performed. This analysis involved modifying or adding joint HEPs for situations where multiple HFEs occurred in the same cut set.

16.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the HRA analysis performed by the reference plant. The review stated that the reference plant's HRA was fairly comprehensive, though there were some deviations from the common practice. The review made note that all of the internal event HFEs were carried into the fire model, along with the Level 2 PRA recovery events. The review identified one area of the reference plant's HRA that could potentially impact the overall fire PRA results. This area deals with MCR abandonment and the potential use of alternate shutdown capability (ASC). The reference plant assumed the ASC is failed and therefore no credit is taken for this capability. This assumption has no significant impact on CDF; however, it significantly impacts LERF. The SNL review expressed that the L3PRA project team may want to further evaluate MCR abandonment scenarios, including crediting use of the ASC, as part of the Level 2 PRA.

16.4 L3-FPRA Approach to Address the Task

The HRA performed and used for the L3-FPRA is discussed in this section and documented in Appendix A. The HFEs were determined based on the L3PRA Level 1 internal events model and evaluation of the L3-FPRA fire scenarios. The Level 1 internal event HFEs that required detailed HRA were determined by solving the fire scenarios with all HFEs set to a probability of 0.9. This identified the important HFEs to be evaluated (i.e., those with the potential to make the largest contribution to fire CDF).

This section discusses those HFEs that are modeled in the L3-FPRA. These HFEs either were already included in the Level 1 internal events model or were introduced as new HFEs in the L3-FPRA to address conditions specific to one or more internal fire scenarios. The new HFEs were identified based on review of the RP-FPRA.

Section 16.4.1 provides a discussion on the development of the independent HFEs that are modeled in the L3-FPRA, while Section 16.4.2 discusses the evaluation of the dependencies between HFEs. The level of dependency between HFEs is assessed whenever multiple HFEs exist in a single cut set.

16.4.1 Identification and Quantification of Independent Fire HFEs

The independent fire HFEs were identified by two methods. The first method involved solving the fire-specific event trees with the L3PRA Level 1 internal event HEPs set to 0.9. This method identified the internal event HFEs that are most likely to impact the fire CDF. These identified HFEs were then replaced with a fire-specific HFE by the use of post-processing rules. The second method identified fire-specific HFEs based on logic modeling and other related information from the RP-FPRA.

Once the independent fire HFEs were identified, an initial quantification was performed using the Scoping Approach as outlined in NUREG-1921 (NRC, 2012). However, the results obtained through the application of the Scoping Approach were determined to be unrealistic and differed in several instances quite drastically from the HEPs obtained in reference plant's fire HRA, as well as from the L3PRA Level 1 internal event HRA (NRC, 2016a). In many cases this was due to the Scoping Approach assigning a failure probability of 1.0 if the HFE would be considered "cognitively complex" due to the plant not responding as expected. In these cases, the Scoping Approach recommends a more detailed analysis; however, a more detailed analysis of all identified HFEs was unfeasible given lack of resources and access to plant personnel. Therefore, it was decided to primarily use the reference plant's fire HRA.

Based on the above, the L3-FPRA HEPs were calculated using the following rules:

1. If available, use already calculated fire HEPs from RP-FPRA documentation, since they are supported by the most detailed analysis.

2. If a RP-FPRA fire HEP is less than the corresponding L3PRA Level 1 internal events HEP for the same action, then use the L3PRA Level 1 internal events HEP.[7,8]

3. If a RP-FPRA fire HEP is not available, use the scoping HEP developed by SNL.

There are two exceptions to Rule 3, as described below:[9]

1. For RCS-XHE-XM-TRIP-FIRE, the HEP is set to 0.33 (the L3PRA Level 1 internal events HEP). After discussions with HRA experts and because the scoping HEP value of 1.0 was considered overly conservative and would have a significant impact on the total fire CDF, it was decided to use the L3PRA Level 1 internal events HEP.

2. For CHG-XHE-NORMAL-FIRE, the HEP is set to 3.2×10^{-4} (the L3PRA Level 1 internal events HEP). After discussions with HRA experts and because the scoping HEP value of 1.0 was considered overly conservative for this action and would have a significant impact on the total fire CDF, it was decided to use the L3PRA Level 1 internal events HEP.

The independent fire HEPs used in the L3-FPRA, their uncertainty distribution information, error factor, and their reference source are provided in Table 16-1.

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
1	1-CAD-XHE-SAFESTBLE-FIRE	OPERATOR FAILS TO DEPRESSURIZE SECONDARY (72HR SAFE/STABLE) - FIRE	7.50E-04	Lognormal	10	L1-IE
2	1-CAD-XHE-SGTR-LT-FIRE	FAILURE TO INITIATE NORMAL COOLDOWN	1.90E-03	Lognormal	5	L1-IE

7 Rule 2 was used because the L3PRA project's HRA team did not identify any reasons why the likelihood of operator error should be lower under fire conditions and to maintain internal (model) consistency.

8 In almost all instances where this rule was applied, the internal events HEP was the one that was reanalyzed as part of the L3PRA Level 1 internal events HRA, which typically resulted in an increase in the HEP as compared to the value used in the reference plant internal events PRA. In these instances, the HEP in the reference plant fire PRA was generally greater than the corresponding HEP in the reference plant internal events PRA. For the few cases where the HEP in the reference plant fire PRA was less than the corresponding HEP in the reference plant internal events PRA, this difference was small.

9 Since the two exceptions listed are the only instances where Rule 3 was applied, ultimately, none of the SNL scoping HEPs were used in the L3-FPRA.

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
		WITH HPI - SGTR, LATE - FIRE				
3	1-CHG-XHE-NORMAL-FIRE	OPERATOR FAILS TO ESTABLISH CHARGING GIVEN A LOSS OF RCP SEAL INJECTION - FIRE	3.20E-04	Lognormal	10	L1-IE
4	1-OA-ALIGNPW-01HR-FIRE	OPERATOR FAILS TO ALIGN ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 1 HR AFTER SBO - FIRE	1.15E-01	Lognormal	5	RP Fire
5	1-OA-ALIGNPW-02HR-FIRE	OPERATOR FAILS TO ALIGN ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 2HR AFTER SBO - FIRE	1.22E-02	Lognormal	5	L1-IE
6	1-OA-ALTAFW----H-FIRE	OPERATOR FAILS TO PROVIDE ADDITIONAL WATER SOURCE FOR LONG TERM AFW - FIRE	1.32E-03	Lognormal	5	RP Fire
7	1-OAB_SI-------H-FIRE	OPERATOR FAILS TO BLEED & FEED - SI present - FIRE	2.35E-02	Lognormal	5	L1-IE
8	1-OAB_TR-------H-FIRE	OPERATOR FAILS TO FEED AND BLEED - TRANSIENT - FIRE	5.80E-02	Lognormal	5	L1-IE
9	1-OAB-SBOACR---H-FIRE	OPERATOR FAILS TO INITIATE FEED AND BLEED - SBO ACR - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
10	1-OAC_AC-------H-FIRE	OPERATOR FAILS TO DEPRESSURIZE FOR LPI- SLOCA HPI FAILED - FIRE	4.38E-03	Lognormal	5	RP Fire
11	1-OAC_NC-------H-FIRE	OPERATOR FAILS TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE	2.19E-03	Lognormal	5	RP Fire
12	1-OA-CCP-ALIGN---H-FIRE	OPERATOR FAILS TO SHIFT FROM NCP TO CCP AFTER LOACCW FOR RCP SL INJ. - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
13	1-OACONTROL--AFW-FIRE	OPERATOR FAILS TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE	7.40E-02	Lognormal	5	RP Fire
14	1-OA-CSISOL----H-FIRE	OPERATOR FAILS TO CLOSE CS SUCTION	2.60E-02	Lognormal	5	RP Fire

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
		FROM THE RWST - FIRE				
15	1-OAD_MLA------H-FIRE	OPERATOR FAILS TO DEPRESS SECONDARY FOR LPI - MLO w HPI failed - FIRE	4.44E-01	Lognormal	3	L1-IE
16	1-OAD_SGR------H-FIRE	OPERATOR FAILS TO DEPRESSURIZE SECONDARY - FIRE	1.45E-03	Lognormal	5	L1-IE
17	1-OA-DEP-SBO---H-FIRE	OPERATOR FAILS TO DEP. SG TO 300 psig IN SBO -local ARV operation - FIRE	2.70E-02	Lognormal	5	L1-IE
18	1-OA-ESFAS-HE1-H-FIRE	OPERATOR FAILS TO START EQUIP ON FAILURE OF ESFAS SIGNAL - FIRE	1.86E-02	Lognormal	5	RP Fire
19	1-OAF_MFW------H-FIRE	OPERATOR FAILS TO ESTABLISH MFW TO SGs - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
20	1-OA-HPR-ACRA--H-FIRE	OPERATOR FAILS TO SWITCH TO HPR - SBO AC recov 21/480 gpm or STKO RV w CCUs - FIRE	1.18E-03	Lognormal	5	L1-IE
21	1-OA-HPRCU-ACR-H-FIRE	OPERATOR FAILS TO ESTABLISH HPR - SBO after ACR 21/182gpm w/o CCUs - FIRE	7.90E-04	Lognormal	10	L1-IE
22	1-OA-HURGXFMR--H-FIRE	OPERATOR FAILS LOCAL CHANGE 120VAC SUPPLY FROM INVRTR TO RGXFMR - FIRE	8.50E-03	Lognormal	5	RP Fire
23	1-OAI_SG-------H-FIRE	OPERATOR FAILS TO ISOLATE RUPTURED SG - FIRE	2.10E-02	Lognormal	5	L1-IE
24	1-OA-IS-ISLACC-H-FIRE	OPERATOR FAILS TO ISOLATE ISLOCA THROUGH ACCW RCP TB COOLING LINE - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
25	1-OA-IS-ISLCP--H-FIRE	OPERATOR FAILS TO LOCATE ISLOCA PATH TO NCP/CCPS SUCTION - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
26	1-OA-IS-ISLLKF-H-FIRE	OPERATOR FAILS TO ISOLATE RCP SEAL LEAK OFF ISOLATION VALVES - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
27	1-OA-IS-ISLRHR-H-FIRE	OPERATOR FAILS TO ISOLATE ISLOCA	1.00E+00	Point Estimate	N/A	RP Fire

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
		THROUGH RHR CL INJ. LINES - FIRE				
28	1-OA-IS-ISLSEALSBO-FIRE	OPERATOR FAILS TO ISOLATE RCP SEAL LINES at LOCAL - ISLOCA w SBO - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
29	1-OA-IS-ISLSI--H-FIRE	OPERATOR FAILS TO ISOLATE ISLOCA PATH through SIS CL OR HL INJ LINES - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
30	1-OA-ISL-MITI--H-FIRE	OPERATOR FAILS TO MITIGATE AND STABILIZE PLANT AFTER SMALL SIZE ISLOCA - FIRE	1.73E-04	Lognormal	10	RP Fire
31	1-OA-ISOLETDOWNH-FIRE	OPERATOR FAILS TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE	7.60E-02	Lognormal	5	RP Fire
32	1-OAISOLSTMTDAFW-FIRE	OPERATOR FAILS TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE	2.70E-02	Lognormal	5	RP Fire
33	1-OAL_LPLL-----H-FIRE	OPERATOR FAILS TO ESTABLISH LOW PRESSURE HOT LEG RECIRC - LLO - FIRE	1.30E-04	Lognormal	10	L1-IE
34	1-OA-LTFB-ACRA-H-FIRE	OPERATOR FAILS TO HPR FOR LONG TERM F&B - SBO after AC recov F&B inj. CCU recov - FIRE	6.00E-04	Lognormal	10	L1-IE
35	1-OA------MANRTH-FIRE	OPERATOR FAILS TO MANUALLY INITIATE A REACTOR TRIP - FIRE	1.90E-03	Lognormal	5	RP Fire
36	1-OA-MANUAL-SI-H-FIRE	OPERATOR FAILS TO MANUALLY INITIATE A SAFETY INJECTION - FIRE	1.20E-03	Lognormal	5	RP Fire
37	1-OAN_SL-------H-FIRE	OPERATOR FAILS TO ESTABLISH NORMAL RHR - SLOCA - FIRE	1.10E-03	Lognormal	5	L1-IE
38	1-OA-N1EBATCHG-H-FIRE	OPERATOR FAILS TO PUT THE STANDBY NON 1E BATTERY CHARGER TO SERVICE - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
39	1-OA-NSCWCT-MV-H-FIRE	OPERATOR FAILS TO LOCALLY OPEN NSCW CT SPRAY MOV NO SI - FIRE	1.38E-02	Lognormal	5	RP Fire
40	1-OA-NSCWFAN---H-FIRE	OPERATOR FAILS TO START NSCW FAN	1.00E+00	Point Estimate	N/A	RP Fire

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
		MANUALLY (PLACE HOLDER) - FIRE				
41	1-OA-OBR-------H-FIRE	OPERATOR FAILS TO ESTABLISH EMERGENCY BORATION - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
42	1-OA-OCR_A-----H-FIRE	OPERATOR FAILS TO STEP INSERT CONTROL RODS - FIRE	1.00E+00	Point Estimate	N/A	L1-IE
43	1-OA-OFC_1-----H-FIRE	OPERATOR FAILS TO CONTINUE TO OPERATE TDAFWP AFTER BAT DEPL- SBO w DEP failed - FIRE	3.00E-01	Lognormal	5	L1-IE
44	1-OA-OFC_2-----H-FIRE	OPERATOR FAILS TO CONTINUE TDAFWP AFTER BAT DEPL. - SBO with DEP success - FIRE	3.00E-01	Lognormal	5	L1-IE
45	1-OA-OLP_ML----H-FIRE	OPERATOR FAILS TO RESTART RHR PUMP FOR LPI MLOCA HPI FAILS DPI SUCC - FIRE	2.83E-02	Lognormal	5	RP Fire
46	1-OA-OLP_SL----H-FIRE	OPERATOR FAILS TO RESTART RHR PUMP FOR LPI SLOCA HPI FAILS DPI SUCCESS - FIRE	1.23E-02	Lognormal	5	RP Fire
47	1-OA-OLP_STOPB-H-FIRE	OPERATOR FAILS TO STOP RHR PUMP WHEN RCS P >300 psig (when CCW not avail.) - FIRE	1.59E-02	Lognormal	5	RP Fire
48	1-OA-OP-PHASE-AH-FIRE	OPERATOR FAILS TO MANUALLY INITIATE PHASE A ISOLATION - FIRE	3.00E-03	Lognormal	5	L1-IE
49	1-OA-ORS-------H-FIRE	OPERATOR FAILS TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO - FIRE	5.73E-02	Lognormal	5	L1-IE
50	1-OA-OSW-------H-FIRE	OPERATOR FAILS TO ESTABLISH 1 NSCW PUMP FOR NSCW PUMP 1 2 3 4 5 OR 6 INITIATOR - FIRE	3.76E-02	Lognormal	5	RP Fire
51	1-OA-PORVBLOCKVH-FIRE	OPERATOR FAILS TO CLOSE PRESSURIZER PORV BLOCK VALVES DURING A FIRE - FIRE	2.10E-02	Lognormal	5	RP Fire

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
52	1-OAR_HPATA----H-FIRE	OPERATOR FAILS TO ESTABLISH HPR DURING ATWT - W CCU SUCC (CS NOT ACTUATED) - FIRE	2.31E-03	Lognormal	5	L1-IE
53	1-OAR_HPATB----H-FIRE	OPERATOR FAILS TO ESTABLISH HPR DURING ATWT - W CCU FAILED (CS ACTUATED) - FIRE	2.31E-03	Lognormal	5	L1-IE
54	1-OAR_HPML-----H-FIRE	OPERATOR FAILS TO ESTABLISH HIGH PRESSURE RECIRCULATION - MLOCA - FIRE	2.31E-03	Lognormal	5	L1-IE
55	1-OAR_HPMSO----H-FIRE	OPERATOR FAILS TO ESTABLISH HPR - RWST MSO - FIRE	1.20E-02	Lognormal	5	RP Fire
56	1-OAR_HPSLA----H-FIRE	OPERATOR FAILS TO ESTABLISH HPR - SLOCA with CCUs available - FIRE	6.00E-04	Lognormal	10	L1-IE
57	1-OAR_HPSLB----H-FIRE	OPERATOR FAILS TO ESTABLISH HPR - SLOCA WITH CCUs NOT AVAILABLE - FIRE	2.31E-03	Lognormal	5	L1-IE
58	1-OAR_LPLL-----H-FIRE	OPERATOR FAILS TO ESTABLISH LOW PRESSURE RECIRC - LLO - FIRE	1.40E-02	Lognormal	5	RP Fire
59	1-OAR_LPML-----H-FIRE	OPERATOR FAILS TO ESTABLISH LPR - MLOCA, HPI FAILED, DEP AND LPI SUCCESS - FIRE	1.50E-03	Lognormal	5	RP Fire
60	1-OAR_LPSL2----H-FIRE	OPERATOR FAILS TO ESTABLISH LPR AFTER DEPRESSURIZATION - SLOCA, CCUs FAILED - FIRE	6.80E-04	Lognormal	10	L1-IE
61	1-OAR_LPSL-----H-FIRE	OPERATOR FAILS TO LPR AFTER DEPRESSURIZATION - SLOCA, RHR FAILED, CCUs AVAILABLE - FIRE	1.10E-03	Lognormal	5	L1-IE
62	1-OAR_LPSLNOHI-H-FIRE	OPERATOR FAILS TO ESTABLISH LPR - SLOCA HPI FAILED DEP for LPI & LPI SUCCESS - FIRE	3.70E-05	Lognormal	10	L1-IE

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
63	1-OAR_LTFB_SLA-H-FIRE	OPERATOR FAILS TO ESTABLISH HPR FOR LONG TERM F&B -SLO with CCUs - FIRE	5.80E-04	Lognormal	10	L1-IE
64	1-OAR_LTFB-TRA-H-FIRE	OPERATOR FAILS TO ESTABLISH. HPR FOR LONG TERM F&B - TRANS CCU avail. - FIRE	6.00E-04	Lognormal	10	L1-IE
65	1-OAR_LTFB-TRB-H-FIRE	OPERATOR FAILS TO ESTABLISH HPR FOR LONG TERM F&B - TRANSIENT with CCU fail - FIRE	2.31E-03	Lognormal	5	L1-IE
66	1-OA-SAGD-CHG--H-FIRE	OPERATOR FAILS TO ESTABLISH SAFETY GRADE CHARGING AFTER LOSINJ IE - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
67	1-OA-START-ACCWH-FIRE	OPERATOR FAILS TO START ACCW PUMP FOR SPECIAL INITIATOR - FIRE	6.40E-02	Lognormal	5	L1-IE
68	1-OA-START-AFW-H-FIRE	OPERATOR FAILS TO MANUALLY START AFW PUMPS IN MCR - FIRE	1.24E-02	Lognormal	5	RP Fire
69	1-OA-SUMPMOV---H-FIRE	OPERATOR FAILS TO OPEN SUMP MOVS FOR RECIRC - auto sig. failed - FIRE	1.80E-03	Lognormal	5	L1-IE
70	1-OAT----------H-FIRE	OPERATOR FAILS TO TERMINATE SI - FIRE	6.00E-04	Lognormal	10	L1-IE
71	1-OAT-ISINJ----H-FIRE	OPERATOR FAILS TO TERMINATE SI AFTER ISINJ INITIATING EVENT - FIRE	3.26E-04	Lognormal	10	RP Fire
72	1-OA-XFER-NON1EH-FIRE	OPERATOR FAILS TO ALIGN NON-1E BUSES GIVEN FAST XFER FAILS - FIRE	1.00E+00	Point Estimate	N/A	RP Fire
73	1-OA-XFER-NON1EH-LT-FIRE	OPERATOR FAILS TO ALIGN NON-1E BUSES GIVEN FAST TRANSFER FAILS - LONG-TERM -FIRE	2.70E-03	Lognormal	5	L1-IE
74	1-RCS-XHE-XM-TRIP-FIRE	OPERATOR FAILS TO TRIP REACTOR COOLANT PUMPS - FIRE	3.30E-01	Lognormal	3	L1-IE
75	1-RFL-XHE-REFILL-LT-FIRE	OPERATOR FAILS TO REFILL RWST LONG-TERM - FIRE	1.00E-04	Lognormal	10	L1-IE

Table 16-1 Independent Fire HEPs

	Name	Description	Independent Fire HEP	Uncertainty Distribution	Error Factor (Note 1)	Reference (Note 2)
76	1-RPS-XHE-XE-NSGNL-FIRE	OPERATOR FAILS TO RESPOND WITH NO RPS SIGNAL PRESENT - FIRE	2.30E-01	Lognormal	5	L1-IE

Notes:

1. The error factors were assigned based on the guidance in Section 19.4.1.
2. L1-IE = L3PRA project Level 1 full power internal event PRA model
   RP Fire = RP-FPRA model

16.4.2 FIRE-HFE Dependency Analysis

As in the Level 1 PRA for internal events, potential dependencies among operator actions within the same sequence must be considered. The specific items of potential importance revolve around an operating staff error that may occur coincidentally with other errors. These combinations of errors may be such that they cannot be treated as random, independent failures.

The approach used to account for potential dependencies among operator actions in the L3-FPRA is described in Section 16.4.2.1. The implementation of this approach is described in Section 16.4.2.2.

16.4.2.1 Dependency Analysis Approach

The ASME/ANS PRA Standard (ASME, 2009) supporting requirement HR-G7 requires the calculation of joint human error probabilities (JHEPs) for multiple human actions that occur within the same cut set. The JHEP calculation should account for timing, common procedures, common instruments, and personnel resources. The reference plant's Full Power Internal Events PRA Notebook provides a discussion on identifying significant operator actions and assigning dependency factors [complete dependence (CD), high dependence (HD), moderate dependence (MD), low dependence (LD), and zero dependence (ZD)] between independent HEPs. Section 16.4.2.2 and Appendix A of this report discuss how the dependencies were determined and the calculation of the JHEPs for the dependent operator actions for the L3-FPRA. The JHEP values were adjusted using the independent HEP values as modified for fire initiators.

To determine the different combinations of HFEs that would show up together within a sequence cut set, all of the HEPs were set to 0.9 and the model was solved at a truncation of 10^{-10}/rcy. This was a low enough truncation using the high HEPs to make sure all of the important combinations would propagate above truncation and the model would solve in a reasonable time. The top 200 cut sets that contained independent HFEs were reviewed to identify all combinations of multiple HFEs existing in a single cut set. The level of dependency to be assigned to each combination of HFEs was then evaluated based on the following factors (which are described in Section 16.4.2.2):

- Same crew
- Common cognitive
- Same time/timing
- Adequate resources
- Same location

The cut set review described above identified 65 different HFE combinations, which were then reviewed to assess the level of dependence between each pair of HFEs in these combinations. Table 16-2 provides a list of the 65 different HFE combinations that were evaluated, and their assigned dependency levels.

SAPHIRE post-processing rules were used to replace the independent HFEs with dependent versions (with modified HEPs), as needed. These rules are designed to search for combinations of HFEs within a single cut set, and then replace the second (or subsequent) independent HFE(s) with a new HFE representing a conditional probability (dependent HFE). For each combination, the analyst identifies the HFE that fails first and determines if the failure of the first HFE can have some influence on (i.e., increase the failure of) the second. If so, then a new conditional probability (HEP) is calculated and used for the second (or subsequent) HFE.

The dependent combinations that were identified via the process discussed above are listed in Table 16-2. The final dependent HEPs used in the L3-FPRA are listed in Table 16-3.

16.4.2.2 Dependency Analysis Implementation

To ensure that all key HFE dependency combinations were identified, the L3-FPRA model was solved by setting the HEPs for applicable HFEs to 0.9.[10] A cut set contribution cutoff of 1×10^{-10} /rcy was used, since this value was judged to be sufficient to prevent any potential HFE combinations that could significantly affect the internal fire CDF from being eliminated from the evaluation prior to the dependency review. This resulted in 212,700 cut sets (with elevated HEPs) with CDFs greater than or equal to 1×10^{-10} /rcy.[11] From these cut sets, the top 200 cut sets that contain HFE combinations (i.e., two or more HFEs) were evaluated. Using a lower cut set CDF screening frequency threshold could add a significant number of additional HFE combinations; however, it is not believed that the additional combinations will result in many new HFE pairs (i.e., most, if not all, will contain HFE pairs already included in this evaluation). From these cut sets, 65 HFE dependency combinations were identified.

After the dependent HFEs were identified, the level of dependency was determined. NUREG-1792 (NRC, 2005) presents guidance on determining the level of dependency between HFEs. NUREG-1792 presents a few specific elements that should be evaluated for determining level of dependency, such as:

- The same crew member(s) are responsible for the actions.

- The actions can be considered to take place relatively close in time such that a common crew mindset may carry over from one action to the next.

[10] HEPs that were currently set to 1.0 (or “TRUE”) were left at that value, while all others were set to 0.9.

[11] The corresponding CDFs for these cut sets without the elevated HEPs applied will be at least an order of magnitude lower even if complete dependency exists, because the first HFE in the combination is independent (i.e., its HEP would not be elevated).

- There are similar plant conditions between the actions, and they are being directed by identical (or nearly so) procedure and cue.

- The actions are performed in the same location and performed in similar ways.

More generally, NUREG-1792 suggests evaluating if the actions have similar performance shaping factors (PSFs) and if there is reason to believe that the crew's interpretation of the need or decision for an action might influence the crew's decisions for actions later in the scenario.

The L3-FPRA model considers these elements in determining the level of dependency and uses essentially the same approach as used for the L3PRA internal event PRA (see later in this section for discussion of how stress was treated differently from the internal event PRA when determining the level of dependence between two HFEs). The approach for determining the dependency level specifically considers the elements outlined in NUREG-1792 of same crew, timing of cues and action, same procedures and cues, and same location. In addition, the approach applied for the fire HRA assesses if enough resources are available for the successful completion of both actions. The combination of all these elements starts to address the more general suggestion by NUREG-1792 to evaluate similar PSFs across the actions. In addition, the evaluation of same procedures and cues for the actions is asserted to address an element termed "common cognitive," referring to the crew having a common mindset or belief in approaching the actions. This "common cognitive" is likely to also influence the crew's decisions for actions later in the scenario. Specifically, the following criteria are evaluated through this approach:

- **Same Crew.** If the actions are assumed to be performed by different crews, the HFEs may be considered to be independent. If the difference in time between the cues for each of the HFEs is greater than the length of the shift (12 hours at the reference plant), a new crew can be assumed to be responding to the cue of the second HFE.

- **Common Cognitive / Same Cues and Procedures.** If the crew can be assumed to be in a common cognitive mindset while responding to both HFEs, complete dependency is assigned. This element is assessed by evaluating if the cue and/or procedure steps being used are essentially identical for the HFEs being evaluated. NUREG-1921 defines a cue as "a change in condition or signal that triggers the need for an action" (pg. A-2); therefore, a cue can be thought of as either an indication, procedure step, or alarm that alerts the crew member to need for a response.

- **Time.** This element assesses the amount of time that is estimated to have elapsed between the cues for each of the HFEs. The options are that the cues occur simultaneously or differ by one of the following intervals:

  $0 <$ Time ≤ 15 minutes
  $15 <$ Time ≤ 30 minutes
  $30 <$ Time ≤ 60 minutes
  Time > 60 minutes

- **Adequate Resources.** This element assesses whether an adequate number of staff is available to support the required actions. For the dependency determination, this assessment is only relevant if the actions are required to be completed during the same time. If staffing is found to be insufficient, complete dependency is assumed.

- **Same Location.** The location refers to the room or general area in which the actions will be executed. If the actions are executed in the same location, a higher level of dependency is typically assessed.

Using this approach, a level of dependency is assigned, as shown in Figure 16-1, of either: complete, high, moderate, low, or zero.

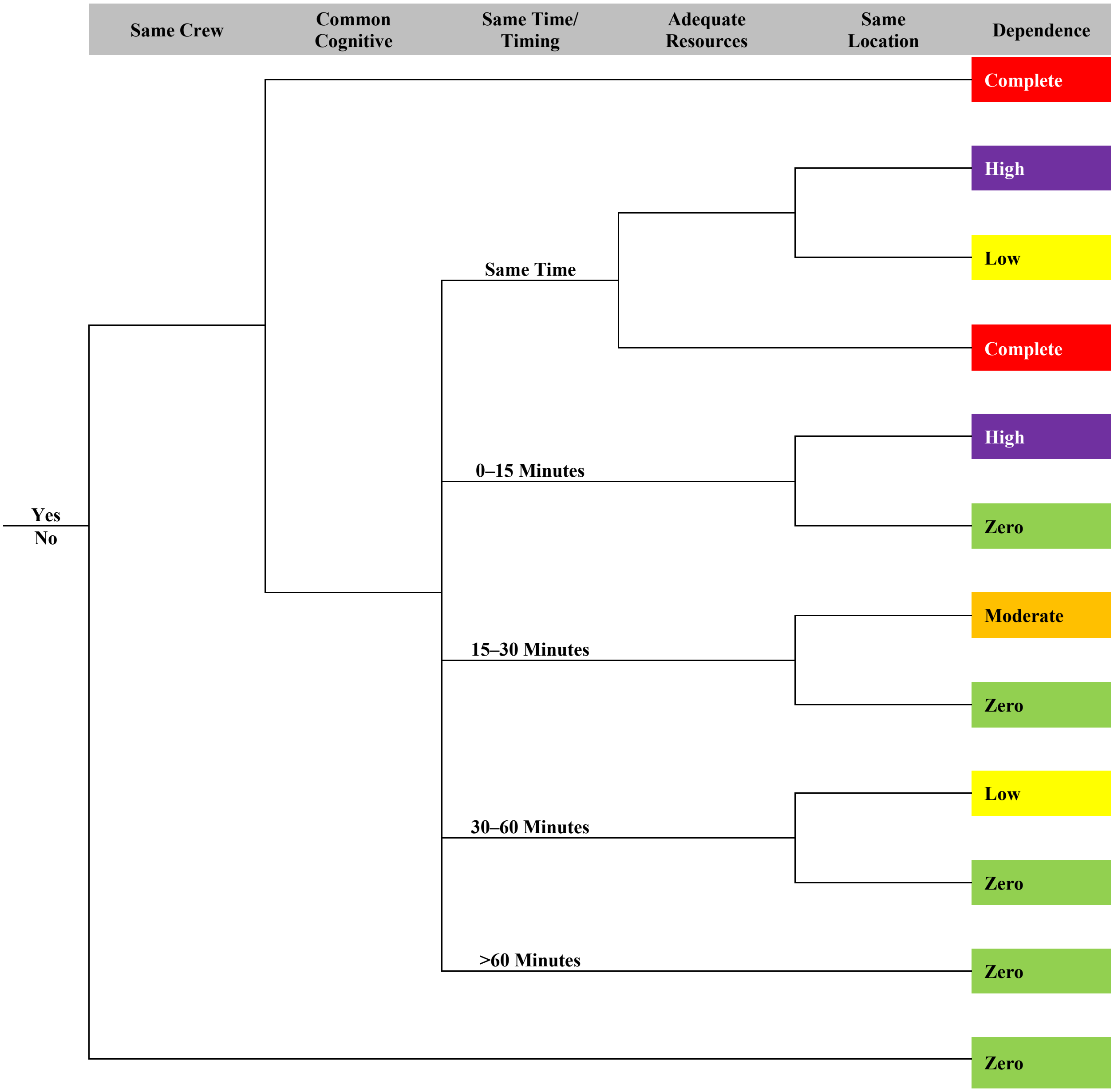

Figure 16-1 L3PRA Modified Dependency Decision Tree

Note, there is one significant difference in how HFE dependency was evaluated in the L3-FPRA as compared to the L3PRA internal event PRA. The EPRI HRA Calculator dependency decision tree includes a node for stress. However, the guidance provided in the EPRI HRA Calculator Version 5.1 User's Manual (EPRI, 2013) for assessing the parameters to determine level of dependency offers no guidance on how to evaluate level of stress for dependency calculations. The L3PRA internal event PRA conservatively used the higher branch (either high or moderate) for the stress node in the decision tree without evaluating stress explicitly. After the HFE dependencies were calculated using this assumption, and incorporated into the L3PRA internal event PRA model, a review of the model results concluded that while this assumption does impact the calculated CDF for the Level 1 internal event PRA, it is not a major contributor. However, it was later realized that this conservative treatment of stress could have a significant impact for subsequent L3PRA models (e.g., the low power and shutdown PRA for internal events).

Upon further discussion of this issue, the HRA team concluded that it seems more appropriate to remove the consideration of stress from the dependency analysis (i.e., assign dependency levels in all cases based on low stress). Although the EPRI HRA Calculator does not offer guidance on how to assess stress for purposes of assessing dependency level, a definition of stress in this context is given in Section 6 of NUREG-1921 (NRC, 2012). NUREG-1921 defines stress (in the context of dependency determination) as, “Stress is a culmination of all other performance shaping factors. These factors may include preceding functional failures and successes, preceding operator errors or successes, the availability of cues and appropriate procedures, workload, environment (i.e., heat, humidity, lighting, atmosphere, and radiation), the requirement and availability of tools or parts, and the accessibility of locations. In general, stress is considered high for loss-of-support-system scenarios or when the operators need to progress to functional restoration or emergency contingency action procedures. The higher the stress level, the higher the dependency level” (pg. 6-6). Based on this definition, the HRA team concluded that the decision to assess stress as low is justified. It is not expected that any of the situations that will be assessed will meet the specific considerations specified in the definition for high stress given in NUREG-1921 (i.e., loss-of-support-system scenarios or when operators must progress to functional restoration or emergency contingency actions).

It was also decided that the revised approach for treating stress in the HFE dependency analysis would be applied for all L3PRA models. However, already completed models (e.g., the Level 1 internal event PRA model) were not modified to incorporate this change. It should be noted that this change to the assessment of dependency levels is consistent with the Level 1 internal event PRA in that stress is still not being evaluated. The difference is that the Level 1 internal event PRA always assumes higher stress and the other L3PRA models always assume lower stress. Optimally, the decision to use the lower stress branches from the HRA Calculator dependency determination tree would have been applied to the whole project; however, the decision to not change already completed models was driven by resource limitations as well as the fact that it is not a major contributor to internal events CDF.

Lastly, initial analysis of the fire PRA quantification results identified a major contribution from cut sets that involve operator failure to trip the reactor coolant pumps (RCPs) following a loss of all RCP seal injection and cooling (RCS-XHE-XM-TRIP-FIRE) and additional HFEs. Per the WOG 2000 RCP seal LOCA model, this action needs to be taken within 13 minutes of the loss of RCP seal cooling. However, the procedure only directs the operators to trip the RCPs after they have spent 10 minutes attempting to restore seal cooling. Further analysis of this HFE under the conditions being considered led to the expectation that this failure would most likely result from the operators failing to take the action within the required time, rather than from failure to recognize the need to take the action. As such, it was assumed that there would be no cognitive connection between this HFE and any subsequent HFEs in the associated cut sets. In addition, this further analysis led to the expectation that the impact of “same location” would be overly conservative in the assessment of potential dependencies between this HFE and any subsequent HFEs. Based on this information, it was decided that zero dependence should be assumed for all HFE pairs that include RCS-XHE-XM-TRIP-FIRE as the first HFE in the pair.

Following the assignment of the dependency level for each dependent HFE, its HEP was recalculated by applying the following dependency formulas given in NUREG/CR-1278 (NRC, 1983):

Dependence Level	Equation
Zero	HEP
Low	(1 + 19 × HEP) / 20
Moderate	(1 + 6 × HEP) / 7
High	(1 + HEP) / 2
Complete	1.0

If more than two HFEs are considered to be dependent in a sequence, the dependency calculation for each successive HFE is calculated based only on the immediately preceding HFE. In other words, if three HFEs termed HFE1, HFE2, and HFE3 were found to be dependent, the calculation of the dependent HEP for HFE2 would be based on the dependency level determined between HFE1 and HFE2. The calculation used to determine the HEP for the dependent HFE3 is based on the dependency level determined between HFE2 and HFE3. This calculation is based on guidance given in NUREG/CR-1278.

Appendix A provides the dependency analysis details for each HFE pair evaluated.

Table 16-2 Fire HFE Dependency Rules for Two or More Operator Actions within a Cut Set

Rule No.	1ST HFE	2nd HFE	Dependency Level	3rd HFE	Dependency Level	4th HFE	Dependency Level
I. Rules for cut sets with 2 HFEs							
1	1-CHG-XHE-NORMAL-FIRE	1-CAD-XHE-SAFESTBLE-FIRE	Zero				
2	1-OA-ALTAFW----H-FIRE	1-CHG-XHE-NORMAL-FIRE	Zero				
3	1-OAR_HPMSO----H-FIRE	1-CAD-XHE-SAFESTBLE-FIRE	Zero				
4	1- OAR_HPSLA----H-FIRE	1-CAD-XHE-SAFESTBLE-FIRE	Zero				
5	1-OAT----------H-FIRE	1-OAR_HPSLA----H-FIRE	Zero				
6	1-OAT----------H-FIRE	1-OAC_NC-------H-FIRE	Moderate				
7	1-OAT----------H-FIRE	1-OAN_SL-------H-FIRE	Zero				
8	1-OAT-ISINJ----H-FIRE	1-OAR_HPSLA----H-FIRE	Zero				
9	1-RCS-XHE-XM-TRIP-FIRE	1-OAR_HPSLA----H-FIRE	Zero				
10	1-RCS-XHE-XM-TRIP-FIRE	1-OAR_LPSL-----H-FIRE	Zero				
11	1-OAR_HPSLA----H-FIRE	1-OA-ALTAFW----H-FIRE	Zero				
12	1-OAR_HPMSO----H-FIRE	1-OA-ALTAFW----H-FIRE	Zero				
13	1-OAC_AC-------H-FIRE	1-OAR_HPSLA----H-FIRE	Zero				
14	1-OAN_SL-------H-FIRE	1-OAR_LPSL-----H-FIRE	Zero				
15	1-OAT-ISINJ----H-FIRE	1-OAN_SL-------H-FIRE	Zero				
16	1-RCS-XHE-XM-TRIP-FIRE	1-OAN_SL-------H-FIRE	Zero				
17	1-OACONTROL--AFW-FIRE	1-OAB_TR-------H-FIRE	Moderate				
18	1-OACONTROL--AFW-FIRE	1-OAR_LTFB-TRA-H-FIRE	Zero				
19	1-OAT-ISINJ----H-FIRE	1-OAC_NC-------H-FIRE	Moderate				
20	1-OA-ISOLETDOWNH-FIRE	1-OAR_LPSL-----H-FIRE	Zero				
21	1-OAISOLSTMTDAFW-FIRE	1-OAR_LTFB-TRA-H-FIRE	Zero				
22	1-OAC_AC-------H-FIRE	1-OAR_HPMSO----H-FIRE	Zero				
23	1-OACONTROL--AFW-FIRE	1-OAR_HPMSO----H-FIRE	Zero				
24	1-OA-START-AFW-H-FIRE	1-OAR_HPMSO----H-FIRE	Zero				
25	1-OA-START-AFW-H-FIRE	1-OAB_TR-------H-FIRE	High				
26	1-OAISOLSTMTDAFW-FIRE	1-OAB_TR-------H-FIRE	Moderate				
27	1-RCS-XHE-XM-TRIP-FIRE	1-OAC_NC-------H-FIRE	Zero				
28	1-OA-ISOLETDOWNH-FIRE	1-OAR_HPSLA----H-FIRE	Zero				

Table 16-2 Fire HFE Dependency Rules for Two or More Operator Actions within a Cut Set (continued)

II. Rules for cut sets with 3 HFEs								
Rule No.	**1ST HFE**	**2nd HFE**	**Dependency Level**	**3rd HFE**	**Dependency Level**			
29	1-OAT----------H-FIRE	1-OAR_LPSL-----H-FIRE	Low	1-OAN_SL-------H-FIRE	Zero			
30	1-OAT----------H-FIRE	1-OAR_HPMSO----H-FIRE	Zero	1-OAC_AC-------H-FIRE	Zero			
31	1-OAT----------H-FIRE	1-OAR_HPSLA----H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Complete			
32	1-OAT----------H-FIRE	1-OA-OLP_SL----H-FIRE	Zero	1-OAR_HPSLA----H-FIRE	Zero			
33	1-OAT----------H-FIRE	1-OAR_HPMSO----H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero			
34	1-OAT----------H-FIRE	1-OAC_AC-------H-FIRE	Moderate	1-OAR_HPSLA----H-FIRE	Zero			
35	1-OAT----------H-FIRE	1-OA-OLP_SL----H-FIRE	Zero	1-OAR_HPMSO----H-FIRE	Zero			
36	1-OACONTROL--AFW-FIRE	1-OA-CSISOL----H-FIRE	High	1-OAR_HPMSO----H-FIRE	Zero			
37	1-OA-START-AFW-H-FIRE	1-OA-CSISOL----H-FIRE	High	1-OAR_HPMSO----H-FIRE	Zero			
38	1-OA-ISOLETDOWNH-FIRE	1-OAC_AC-------H-FIRE	Low	1-OAR_HPMSO----H-FIRE	Zero			
39	1-OAT-ISINJ----H-FIRE	1-OAC_AC-------H-FIRE	Moderate	1-OAR_HPMSO----H-FIRE	Zero			
40	1-OA-CSISOL----H-FIRE	1-OAC_AC-------H-FIRE	Low	1-OAR_HPMSO----H-FIRE	Zero			
41	1-RCS-XHE-XM-TRIP-FIRE	1-OAC_AC-------H-FIRE	Zero	1-OAR_HPMSO----H-FIRE	Zero			
42	1-OAT-ISINJ----H-FIRE	1-OA-OLP_SL----H-FIRE	Zero	1-OAR_HPMSO----H-FIRE	Zero			
43	1-OAT-ISINJ----H-FIRE	1-OAR_HPMSO----H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero			
44	1-OAT-ISINJ----H-FIRE	1-OAC_NC-------H-FIRE	Moderate	1-OAR_HPSLA----H-FIRE	Zero			
45	1-OAT-ISINJ----H-FIRE	1-OA-OLP_SL----H-FIRE	Zero	1-OAR_HPSLA----H-FIRE	Zero			
46	1-RCS-XHE-XM-TRIP-FIRE	1-OAC_AC-------H-FIRE	Zero	1-OAR_HPSLA----H-FIRE	Zero			
47	1-RCS-XHE-XM-TRIP-FIRE	1-OAC_NC-------H-FIRE	Zero	1-OAR_HPSLA----H-FIRE	Zero			
48	1-OAT-ISINJ----H-FIRE	1-OAR_HPSLA----H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Complete			
49	1-OA-ISOLETDOWNH-FIRE	1-OAC_AC-------H-FIRE	Low	1-OAR_HPSLA----H-FIRE	Zero			
50	1-RCS-XHE-XM-TRIP-FIRE	1-OAN_SL-------H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero			
51	1-OA-ISOLETDOWNH-FIRE	1-OAN_SL-------H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero			
52	1-OAT-ISINJ----H-FIRE	1-OAN_SL-------H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero			
53	1-OACONTROL--AFW-FIRE	1-OAISOLSTMTDAFW-FIRE	Complete	1-OAR_LTFB-TRA-H-FIRE	Zero			
54	1-OACONTROL--AFW-FIRE	1-OAISOLSTMTDAFW-FIRE	Complete	1-OAB_TR-------H-FIRE	Moderate			

Table 16-2 Fire HFE Dependency Rules for Two or More Operator Actions within a Cut Set (continued)

III. Rules for cut sets with 4 HFEs							
Rule No.	**1ST HFE**	**2nd HFE**	**Dependency Level**	**3rd HFE**	**Dependency Level**	**4th HFE**	**Dependency Level**
55	1-OAT----------H-FIRE	1-OA-CSISOL----H-FIRE	High	1-OAC_AC-------H-FIRE	Low	1-OAR_HPMSO----H-FIRE	Zero
56	1-OAT----------H-FIRE	1-OAISOLSTMTDAFW-FIRE	High	1-OAN_SL-------H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero
57	1-OAT----------H-FIRE	1-OAISOLSTMTDAFW-FIRE	High	1-OAC_NC-------H-FIRE	Low	1-OAR_HPSLA----H-FIRE	Zero
58	1-OACONTROL--AFW-FIRE	1-OAISOLSTMTDAFW-FIRE	Complete	1-OAT----------H-FIRE	High	1-OAR_LPSL-----H-FIRE	Low
59	1-OA-CSISOL----H-FIRE	1-OA-ISOLETDOWNH-FIRE	High	1-OAC_AC-------H-FIRE	Low	1-OAR_HPMSO----H-FIRE	Zero
60	1-OA-CSISOL----H-FIRE	1-OA-ISOLETDOWNH-FIRE	High	1-OA-OLP_SL----H-FIRE	Zero	1-OAR_HPMSO----H-FIRE	Zero
61	1-RCS-XHE-XM-TRIP-FIRE	1-OA-ESFAS-HE1-H-FIRE	Zero	1-OAC_AC-------H-FIRE	Low	1-OAR_HPMSO----H-FIRE	Zero
62	1-OA-CSISOL----H-FIRE	1-OAT-ISINJ----H-FIRE	High	1-OAC_AC-------H-FIRE	Moderate	1-OAR_HPMSO----H-FIRE	Zero
63	1-OA-CSISOL----H-FIRE	1-OA-ISOLETDOWNH-FIRE	High	1-OAR_HPMSO----H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Zero

IV. Rules for cut sets with 5 HFEs									
Rule No.	**1ST HFE**	**2nd HFE**	**Depend. Level**	**3rd HFE**	**Depend. Level**	**4th HFE**	**Depend. Level**	**5th HFE**	**Depend. Level**
64	1-OACONTROL--AFW-FIRE	1-OAISOLSTMTDAFW-FIRE	Complete	1-OAC_NC-------H-FIRE	Low	1-OAT----------H-FIRE	Moderate	1-OAR_HPSLA----H-FIRE	Zero
65	1-OACONTROL--AFW-FIRE	1-OAISOLSTMTDAFW-FIRE	Complete	1-OAN_SL-------H-FIRE	Zero	1-OAT----------H-FIRE	Zero	1-OAR_LPSL-----H-FIRE	Low

Table 16-3 Dependent Fire HFEs

	Name	Description	Dependent FIRE-HEP	Error Factor (Note 1)	Dependency Comment
1	1-OA-CSISOL----H-FIRE-HD	OPERATORS FAIL TO CLOSE CS SUCTION FROM THE RWST - FIRE (HIGH DEPENDENCY)	5.13E-01	2	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 8 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
2	1-OA-ISOLETDOWNH-FIRE-HD	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE (HIGH DEPENDENCY)	5.10E-01	2	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 10 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
3	1-OAB_TR-------H-FIRE-HD	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (HIGH DEPENDENCY)	5.29E-01	2	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 8 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
4	1-OAB_TR-------H-FIRE-MD	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)	1.93E-01	5	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 17 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
5	1-OAC_AC-------H-FIRE-LD	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED (LOW DEPENDENCY)	5.42E-02	5	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 30 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
6	1-OAC_AC-------H-FIRE-MD	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED (MODERATE DEPENDENCY)	1.47E-01	5	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 35 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
7	1-OAC_NC-------H-FIRE-LD	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE (LOW DEPENDENCY)	5.21E-02	5	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be about 1 hour between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
8	1-OAC_NC-------H-FIRE-MD	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE (MODERATE DEPENDENCY)	1.45E-01	5	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 35 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
9	1-OAISOLSTMTDAFW-FIRE-HD	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (HIGH DEPENDENCY)	5.12E-01	2	The same crew may still be on shift, and the actions share the same cue as well as the same timing.
10	1-OAISOLSTMTDAFW-FIRE-CD	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)	1.00E+00	N/A	The same crew may still be on shift, and the actions share the same cue as well as the same timing.

Table 16-3 Dependent Fire HFEs (continued)

	Name	Description	Dependent FIRE-HEP	Error Factor (Note 1)	Dependency Comment
11	1-OAR_LPSL-----H-FIRE-CD	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION (COMPLETE DEPENDENCY)	1.00E+00	N/A	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be less than 44 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
12	1-OAR_LPSL-----H-FIRE-LD	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION (LOW DEPENDENCY)	5.10E-02	5	The same crew may still be on shift; however, it is different cognitive procedure step and there is greater than 1 hour expected between the cues.
13	1-OAT-ISINJ----H-FIRE-HD	OPERATORS FAIL TO TERMINATE SI AFTER ISINJ INITIATING EVENT - FIRE (COMPLETE DEPENDENCY)	5.00E-01	2.1	The same crew may still be on shift; however, it is different cognitive procedure step. There is expected to be 5 minutes between the cues for the two HFEs. Also, both actions are expected to be done within the MCR.
14	1-OAT----------H-FIRE-HD	OPERATORS FAIL TO TERMINATE SI - FIRE (HIGH DEPENDENCY)	5.00E-01	2.1	The same crew may still be on shift; however, it is different cognitive procedure step and there is greater than 1 hour expected between the cues.

Notes:

1. The error factors were assigned based on the guidance in Section 19.4.1.

17 TASK 13 – SEISMIC-FIRE INTERACTION ANALYSIS

17.1 Objective of the Task

The objective of Task 13 is to qualitatively evaluate seismic-induced fires to verify that they are of low risk significance. If this cannot be verified, then a quantitative assessment should be performed.

17.2 Reference Plant Work Performed on the Task

The reference plant performed a qualitative assessment of seismic-induced fires and the documentation discusses the use of the Individual Plant Examination for External Events (IPEEE) as the starting point and then verified its conclusions through walkdowns. The documentation also highlights the assumptions about seismically induced component failures and potential ignition sources. The documentation stated that seismically induced fires were not a concern at the reference plant for a high confidence of a low probability of failure (HCLPF) capacity of 0.3g peak ground acceleration (g is the acceleration due to gravity, that is, 9.81 m/s^2).

17.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the information provided by the reference plant for the seismic-fire interaction. This review noted that the evaluation performed by the reference plant was strictly qualitative in nature and consistent with typical practices. SNL also noted that the seismic-fire interaction should be outside the scope of the fire PRA that is currently being developed, since there is no basis in modern fire PRA methodology or in the RP-FPRA documentation to quantify these scenarios.

17.4 L3-FPRA Approach to Address the Task

A qualitative assessment of seismic-fire interaction issues is a fire PRA requirement in the ASME/ANS standard (ASME, 2009). This high-level requirement and its supporting requirements correspond to the qualitative assessments in Task 13 of NUREG/CR-6850 (NRC, 2005). The reference plant commissioned a qualitative assessment of potential seismic-fire interaction issues for the RP-FPRA.

Review of the RP-FPRA documentation by the NRC staff identified that the qualitative assessment in the RP-FPRA addressed all supporting requirements of the ASME/ANS standard and was primarily informed by the reference plant IPEEEs. The NRC review also noted that for specific requirements, the reference plant's assessment also was informed by the review of the plant abnormal operating procedures for a seismic event, plant fire procedures, and plant fire training program procedures.

The reference plant's qualitative assessment concluded that there are no seismic-fire interaction issue concerns at the reference plant for a HCLPF capacity of 0.3g peak ground acceleration. The qualitative assessment for the RP-FPRA was peer reviewed and there are no unresolved facts or observations from that peer review.

The L3-FPRA did not further assess potential seismic-fire interaction issues and does not model or quantify those interaction scenarios.

18 TASK 14 – FIRE RISK QUANTIFICATION

18.1 Objective of the Task

The objective of Task 14 is to quantify the fire PRA to obtain the final fire risk results in terms of CDF and LERF.

18.2 Reference Plant Work Performed on the Task

The reference plant developed and documented a fire PRA model based on the guidance in NUREG/CR-6850. The RP-FPRA documentation discusses the use of flag sets, initiating event treatment, fire sequences being overlayed on the full power internal events model, HRA dependency, and treatment of cable failures, that are all part of the quantification process. The RP-FPRA documentation provides the overall results of the RP-FPRA; that is, the CDF, LERF, importance measures, and dominant contributors to CDF and LERF (in terms of event tree sequences and cut sets) for each unit.

18.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed this task and provided some observations related to main control room (MCR) abandonment, dual unit results, multi-compartment scenarios, and LOOP scenarios. The observation for MCR abandonment scenarios discussed the reference plant's assumption that the full room was damaged and a conditional core damage probability (CCDP) of 1.0 was assigned. These scenarios do not credit the alternate shutdown capability because of the potential spurious operation of the equipment given a fire. The review stated that if a CCDP of 0.1 were used in lieu of 1.0, the LERF would be reduced by 50 percent.

Additional observations from the review include:

- Dual-unit and multi-compartment scenarios have negligible impact on overall results.
- Fire-induced LOOP scenarios are major contributors to the overall results (in part due to the assumption of no offsite power recovery).

The review recommended that the L3PRA project team re-visit some of the conservative assumptions in the RP-FPRA.

18.4 L3-FPRA Approach to Address the Task

The L3-FPRA model was developed using the information provided by the reference plant's CAFTA and FRANX files, as documented in Sections 9 and 15 of this report. The developed L3-FPRA model uses SAPHIRE (SAPHIRE, 2017) to quantify the identified fire scenarios. SAPHIRE is a PC-based software for creating and quantifying fault trees and event trees. The event tree linking (which generates the fire scenario event tree accident sequences) and quantification performed by SAPHIRE produces a CDF estimate and a listing of dominant cut sets and dominant accident sequences. Note, since the L3PRA project includes a full Level 2 PRA for internal fires, no attempt was made to quantify LERF.

Section 18.4.1 describes the CDF quantification process using SAPHIRE. The key results are provided in Section 18.4.2 and dominant fire event tree accident sequence and cut set results are provided in Section 18.4.3. Sections 18.4.4 and 18.4.5 provide important failure events and key insights, respectively. Sections 18.4.6 and 18.4.7 provide some information on limitations associated with the fire mapping process and the model quantification truncation limits, respectively. A brief comparison to fire PRA results for other similar plants is provided in Section 18.4.8.

18.4.1 CDF Quantification Process

As mentioned above, SAPHIRE uses event trees and fault trees to generate event tree accident sequences and their corresponding cut sets. Event tree accident sequences are created during the event tree linking process using the event tree logic and also event tree linkage rules. Event tree linkage rules allow the user to:

- Replace one or more event tree top events with a different top event based on the logical conditions defined by the rule.
- Assign flag sets to the event tree accident sequences based on the logical conditions defined by the rule. The flag sets are used to set up the condition of the fire by adjusting the logic structure of the top events and failing components. The components that are identified as being affected by the specific fire scenario being analyzed are set to a guaranteed failure (TRUE).
- Assign event tree accident sequence end states based on the type of initiator being analyzed.

In addition to the above processing, fault tree flag sets are used to “activate” or “deactivate” portions of a fault tree on an event tree accident sequence-by-sequence basis. “House events” are used to trigger these modifications to fault trees. The fault tree flag sets were used to eliminate multiple fault tree models. For example, flag sets are used on systems in the LOOP and station blackout (SBO) event trees where AC power dependency changes from offsite to onsite power then back to offsite power. By using the fault tree flag sets, only one logic model is required to handle the change in AC power dependency.

Post-processing rules are used by SAPHIRE to perform two basic functions (and discussed in Section 9.4.3.2):

- The post-processing rules remove combinations of test and maintenance events that are disallowed by the plant Technical Specifications. The specific test and maintenance combinations to be removed are identical to those contained in the Level 1 internal event PRA.
- The post-processing rules apply system hardware recovery by appending recovery events to component failure events that are considered recoverable. The L3-FPRA model is similar to most full scope PRAs in that nominal recovery of hardware failures is not generally credited. There are some exceptions. For example, LOOP and SBO models consider recovery of offsite AC power in detail. For some fire scenarios, post-processing rules were used to apply the alternate source of offsite power to cut sets that were created due to the condition of a spurious actuation or consequential LOCA and a LOOP. These conditions did not transfer through the LOOPPC event tree; therefore,

rules were created to appropriately apply recovery (as discussed earlier in Section 9.4.3.2.1).

18.4.2 L3-FPRA Plant CDF Results

The L3-FPRA CDF at power is 6.14×10^{-5} per reactor critical year (/rcy) spread over 210 evaluated fire scenarios. Table 18-1 lists all 210 L3-FPRA fire scenarios and they are sorted from highest to lowest CDF. There are 31 fire scenarios that contribute 1 percent or more to the overall CDF and have a cumulative CDF of 3.43×10^{-5}/rcy (55.9 percent of total fire CDF). From Table 18-1, the top 10 fire scenarios contribute 29.0 percent of the total fire CDF (1.78×10^{-5}/rcy).

The results listed in Table 18-1 are from SAPHIRE after all the fire scenarios have been analyzed and then gathered into a single CD-FIRE end state. Gathering all of the fire scenario cut sets into a single end state allows for Boolean algebra reduction to be performed as if all of the cut sets within a single fire scenario are from a single top fault tree. This process is necessary since proper (complete) handling of success terms cannot be accounted for in SAPHIRE with very complex models. The end state Boolean reduction is particularly important because, as discussed in Section 9, the fire scenarios are processed through eight Level 1 event trees, resulting in the generation of many non-minimal cut sets.

With the non-minimal cut sets removed from within the end state, the frequency for each fire scenario is calculated using the minimal cut set upper bound approximation. Each fire scenario is assumed to be one single group of cut sets when the quantification is performed (all individual event tree sequence boundaries have been removed). Therefore, a single fire frequency is calculated for each fire scenario and these individual fire scenario frequencies are summed together to obtain the overall plant fire CDF.

As discussed in Section 9, the 3,306 RP-FPRA fire sequences were mapped into 210 L3-FPRA fire scenarios. The NRC staff and its contractors performed several comparisons to confirm that the mapping approach did not unduly inflate the CDF for the binned fire scenarios.

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-A105-JY_AX	MCR Panel - AMSAC 11626Q5AMS Fire	3.98E-05	6.73E-02	2.68E-06	4.4%	1	O
IE-FRI-1098-JD_B1	Train B Shutdown Panel 1-1605-P5-SDB Fire - No Spray	2.39E-04	8.86E-03	2.12E-06	3.5%	1	O
IE-FRI-A105-JY_P2	MCB Panel QMCB A1 Fire – NSCW	4.41E-06	4.72E-01	2.08E-06	3.4%	1	O
IE-FRI-1091-J8_B100	4.16 kV AC Swgr 1AA02 CUB. 00 Fire	7.77E-05	2.28E-02	1.77E-06	2.9%	8	G
IE-FRI-1092-J9_C104	4.16 kV AC Swgr 1BA03 Cub. 04 Fire	4.51E-05	3.70E-02	1.67E-06	2.7%	8	G
IE-FRI-1092-J9_C204	4.16 kV AC Swgr 1BA03 Cub. 04 Fire - No Target Damage	9.67E-05	1.60E-02	1.55E-06	2.5%	3	G
IE-FRI-1091-J8_B104	4.16 kV AC Swgr 1AA02 CUB. 04 Fire	3.50E-04	4.43E-03	1.55E-06	2.5%	36	G
IE-FRI-1103-J8_B1	Train A Shutdown Panel 1-1605-P5-SDA Fire - No Spray	2.39E-04	6.18E-03	1.48E-06	2.4%	1	O
IE-FRI-1146-VF_TR01_RR	Bounding Transient	2.76E-03	5.29E-04	1.46E-06	2.4%	12	RR
IE-FRI-1078A-IL_G_RR	125 V DC Panel 1AD11 Fire	9.10E-04	1.58E-03	1.44E-06	2.3%	30	RR
IE-FRI-1094-KQ_B1	U1 Isolating Auxiliary Relay Cabinet 1ACPAR6 Fire	7.96E-05	1.52E-02	1.21E-06	2.0%	1	O
IE-FRI-1091-J8_B200	4.16 kV AC Swgr 1AA02 CUB. 00 Fire - No Target Damage	7.22E-05	1.67E-02	1.20E-06	2.0%	3	G
IE-FRI-2080-M9_H1	480 V AC MCC 2NBR Fire	1.20E-03	9.56E-04	1.15E-06	1.9%	13	G
IE-FRI-1075-I8_C01	480 V AC Swgr 1AB05 Fire	1.40E-04	6.81E-03	9.51E-07	1.6%	1	O
IE-FRI-1140B-S1_B	Elevation 171 - North	5.43E-04	1.71E-03	9.31E-07	1.5%	1	O
IE-FRI-1121-KG_E1	U1 Isolating Auxiliary Relay Cabinet 1BCPAR7 Fire	5.97E-05	1.54E-02	9.22E-07	1.5%	1	O
IE-FRI-1091-J8_C0	Train A Safety Features Sequencer Cabinet 1ACPSQ1	1.99E-04	4.09E-03	8.15E-07	1.3%	1	O
IE-FRI-1120-KH_C4	U1 CSR B Term Cabinet 1BCPT04 Fire - Up to Tray 3	1.06E-05	7.68E-02	8.13E-07	1.3%	4	G
IE-FRI-1091-J8_B212	4.16 kV AC Swgr 1AA02 CUB. 12 Fire - No Target Damage	4.97E-04	1.60E-03	7.94E-07	1.3%	21	G
IE-FRI-1075-I8_F01	480 V AC Swgr 1AB04 Fire	1.40E-04	5.47E-03	7.63E-07	1.2%	1	O
IE-FRI-1120-KH_C6	U1 CSR B Term Cabinet 1BCPT04 Fire - Suppression	1.48E-06	4.80E-01	7.11E-07	1.2%	12	G
IE-FRI-1044-D2_B0	480 V AC MCC 1ABB Fire	4.46E-04	1.56E-03	6.96E-07	1.1%	1	O
IE-FRI-1157A-V3_B1	AFW Train C Turbine Driven Pump Fire	1.96E-05	3.33E-02	6.54E-07	1.1%	3	G
IE-FRI-1092-J9_E0	Train B Safety Features Sequencer Cabinet 1-1821-U	1.16E-04	5.41E-03	6.28E-07	1.0%	1	O

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1075-I8_E1	480 V AC MCC 1ABC Fire	1.12E-04	5.62E-03	6.27E-07	1.0%	1	O
IE-FRI-1140B-S1_E	Elevation 185 – North	8.61E-05	7.23E-03	6.22E-07	1.0%	1	O
IE-FRI-1133B-KK_H0	U1 2A Protection Set II Fire	1.98E-05	3.13E-02	6.21E-07	1.0%	1	O
IE-FRI-1092-J9_C113	4.16 kV AC Swgr 1BA03 Cub. 13 Fire	1.13E-04	5.37E-03	6.06E-07	1.0%	20	G
IE-FRI-1074-ID_E	125 V DC MCC 1CD1M Fire	1.99E-04	2.98E-03	5.94E-07	1.0%	1	O
IE-FRI-1140B-S1_H	Elevation 171 – South	1.48E-04	3.97E-03	5.88E-07	1.0%	1	O
IE-FRI-1080-IS_K2	480 V AC Swgr 1NB09 Fire - Target Damage	3.49E-04	1.68E-03	5.86E-07	1.0%	3	G
IE-FRI-1530_A_RR	Base Scenario	5.28E-02	1.09E-05	5.76E-07	0.9%	14	RR
IE-FRI-1092-J9_C100	4.16 kV AC Swgr 1BA03 Cub. 00 Fire	1.01E-04	5.53E-03	5.61E-07	0.9%	18	G
IE-FRI-1023-B6_TR01	Bounding Transient	8.57E-05	6.20E-03	5.31E-07	0.9%	1	O
IE-FRI-1044-D2_B1	480 V AC MCC 1ABB Fire	1.12E-04	4.70E-03	5.25E-07	0.9%	1	O
IE-FRI-A105-JY_S2	MCB Panel QMCB B1 Fire - FW PT	8.81E-06	5.94E-02	5.24E-07	0.9%	2	G
IE-FRI-AHVSWYD_E	Main Control Panel Fire Results in Loss of Both Of	5.60E-04	9.23E-04	5.17E-07	0.8%	2	G
IE-FRI-1162-T2_B	EDG 1B FIRE/BOUNDING TRANSIENT	3.02E-03	1.64E-04	4.97E-07	0.8%	1	O
IE-FRI-1093-JA_TR04	Transient Along West Wall	5.84E-06	8.18E-02	4.78E-07	0.8%	1	O
IE-FRI-1161-T1_B	EDG 1A FIRE/BOUNDING TRANSIENT	3.02E-03	1.55E-04	4.69E-07	0.8%	1	O
IE-FRI-1092-J9_C221	4.16 kV AC Swgr 1BA03 Cub. 21 Fire - No Target Damage	6.45E-04	7.25E-04	4.67E-07	0.8%	20	G
IE-FRI-1043-D1_B0	480 V AC MCC 1BBB Fire	5.58E-04	8.08E-04	4.51E-07	0.7%	2	G
IE-FRI-2115-JZ_A_RR	Base Scenario	4.19E-02	1.07E-05	4.49E-07	0.7%	25	RR
IE-FRI-1800_A_RR	Base Scenario	4.12E-02	1.07E-05	4.40E-07	0.7%	57	RR
IE-FRI-1512_K2	13.8 kVAC Swgr - 1NAA	2.52E-04	1.65E-03	4.16E-07	0.7%	2	G
IE-FRI-1091-J8_E0	Plant Safety Monitoring System PSMS Cabinet RPUA1	1.84E-05	2.26E-02	4.16E-07	0.7%	2	G
IE-FRI-1120-KH_C3	U1 CSR B Term Cabinet 1BCPT04 Fire - Up to Tray 2	1.40E-05	2.87E-02	4.02E-07	0.7%	2	G
IE-FRI-A105-JY_AR	MCR Panel 1NCQARB Fire	7.96E-05	4.88E-03	3.88E-07	0.6%	2	G
IE-FRI-1120-KH_G4	U1 CSR B Term Cabinet 1BCPT12 Fire - Up to Tray 3	1.98E-05	1.90E-02	3.76E-07	0.6%	4	G

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1120-KH_E4	U1 CSR B Term Cabinet 1BCPT08 Fire - Up to Tray 3	8.12E-06	4.53E-02	3.68E-07	0.6%	3	G
IE-FRI-1160A-V8_C_RR	NSCW Train A Pump 1 Motor Fire	7.78E-04	4.72E-04	3.67E-07	0.6%	11	RR
IE-FRI-1095-JC_F4	U1 CSR A Term Cabinet 11601U3T09 Fire - Up to Tray	2.39E-05	1.52E-02	3.63E-07	0.6%	7	G
IE-FRI-1075-I8_D01	1000 kVA Transformer 1AB05X Fire - First Tray	5.40E-05	6.60E-03	3.57E-07	0.6%	1	O
IE-FRI-2050-D4_A_RR	Base Scenario	3.29E-02	1.07E-05	3.50E-07	0.6%	93	RR
IE-FRI-1507_B1	MAIN FEEDWATER PUMP FIRE - OIL FIRE	4.56E-06	7.64E-02	3.48E-07	0.6%	1	O
IE-FRI-1095-JC_B8	U1 CSR A Term Cabinet 11601U3T01 Fire - Suppression	4.46E-07	7.63E-01	3.40E-07	0.6%	14	G
IE-FRI-1506_JB1	Junction Box 1NQJB6012	1.00E-04	3.31E-03	3.31E-07	0.5%	2	G
IE-FRI-1097-JJ_TR01	Transient - Small North Wall	5.06E-06	6.47E-02	3.27E-07	0.5%	1	O
IE-FRI-1133B-KK_E0	U1 PSMS RPU-B2 1-1625-D5-004 Fire	9.30E-05	3.48E-03	3.23E-07	0.5%	4	G
IE-FRI-1163-T3_A_RR	Base Scenario	2.76E-03	1.14E-04	3.15E-07	0.5%	6	RR
IE-FRI-1080-IS_H2	480 V AC Swgr 1NBL1 Fire - Target Damage	1.85E-04	1.68E-03	3.10E-07	0.5%	1	O
IE-FRI-2091-N4_B100	4.16 kV AC Swgr 2AA02 Cub. 00 Fire	3.41E-04	9.09E-04	3.10E-07	0.5%	114	RR
IE-FRI-1099-J5_A_RR	Base Scenario	9.19E-04	3.32E-04	3.05E-07	0.5%	13	RR
IE-FRI-1120-KH_K3	U1 CSR B Term Cabinet 1BCPT20 Fire - Up to Tray 2	2.96E-05	1.02E-02	3.02E-07	0.5%	4	G
IE-FRI-1508_TR01	TRANSIENT AT CHASE	3.90E-06	7.69E-02	3.00E-07	0.5%	1	O
IE-FRI-1121-KG_B1	U1 Isolating Auxiliary Relay Cabinet 1BCPAR3 Fire	1.19E-04	2.51E-03	2.98E-07	0.5%	1	O
IE-FRI-1120-KH_H2	U1 CSR B Term Cabinet 1BCPT14 Fire - First Tray	1.98E-05	1.44E-02	2.85E-07	0.5%	4	G
IE-FRI-1095-JC_G3	U1 CSR A Term Cabinet 11601U3T11 Fire - Up to Tray	1.88E-05	1.48E-02	2.78E-07	0.5%	3	G
IE-FRI-1506_B1	MAIN FEEDWATER PUMP FIRE - OIL FIRE	4.56E-06	5.95E-02	2.71E-07	0.4%	1	O
IE-FRI-1095-JC_J3	U1 CSR A Term Cabinet 11601U3T17 Fire - Up to Tray	7.95E-05	3.40E-03	2.71E-07	0.4%	15	G
IE-FRI-A105-JY_S3	MCB Panel QMCB B1 Fire - AFW	4.41E-06	5.94E-02	2.62E-07	0.4%	1	O
IE-FRI-A105-JY_L	MCR Panel QPCP Fire	7.96E-05	3.24E-03	2.58E-07	0.4%	2	G
IE-FRI-1120-KH_M4	U1 CSR B Term Cabinet 1NCPT24 Fire - Up to Tray 3	8.12E-06	3.02E-02	2.45E-07	0.4%	3	G
IE-FRI-1188-VH_A_RR	Base Scenario	6.57E-03	3.72E-05	2.44E-07	0.4%	13	RR
IE-FRI-1074-ID_B1	480 V AC MCC 1NBS Fire	9.56E-05	2.52E-03	2.41E-07	0.4%	1	O

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1176-K1_TR05	Transient - Small North Wall	5.06E-06	4.61E-02	2.33E-07	0.4%	1	O
IE-FRI-A105-JY_M	MCR Panel QPP1 Fire	3.98E-05	5.83E-03	2.32E-07	0.4%	1	O
IE-FRI-1095-JC_E3	U1 CSR A Term Cabinet 11601U3T07 Fire - Up to Tray	2.26E-05	1.01E-02	2.29E-07	0.4%	5	G
IE-FRI-1095-JC_G1	U1 CSR A Term Cabinet 11601U3T11 Fire - Panel Only	4.24E-05	5.29E-03	2.24E-07	0.4%	11	G
IE-FRI-1011A-CE_TR01	Bounding Transient	4.25E-05	5.16E-03	2.19E-07	0.4%	1	O
IE-FRI-1094-KQ_H1	U1 Isolating Auxiliary Relay Cabinet 1ACPAR2 Fire	5.31E-05	4.08E-03	2.17E-07	0.4%	1	O
IE-FRI-1120-KH_M5	U1 CSR B Term Cabinet 1NCPT24 Fire - Full ZOI	4.94E-06	4.26E-02	2.10E-07	0.3%	2	G
IE-FRI-1042B-I1_TR03	Transient Level C	3.22E-05	6.52E-03	2.10E-07	0.3%	1	O
IE-FRI-1002-AB_B0	480 V AC Switchgear 1AB15 Fire - HEAF	4.21E-04	4.89E-04	2.06E-07	0.3%	6	G
IE-FRI-A105-JY_Q1	MCB Panel QMCB A2 Fire - RHR	4.41E-06	4.42E-02	1.95E-07	0.3%	1	O
IE-FRI-1048-DC_TR01	Bounding Transient	3.78E-05	5.15E-03	1.95E-07	0.3%	1	O
IE-FRI-A105-JY_AV	MCR Panel 1ACQSTA Fire	3.98E-05	4.85E-03	1.93E-07	0.3%	1	O
IE-FRI-1016-AV_A_RR	Base Scenario	1.40E-02	1.36E-05	1.91E-07	0.3%	326	RR
IE-FRI-1120-KH_E5	U1 CSR B Term Cabinet 1BCPT08 Fire - Full ZOI	2.47E-06	7.63E-02	1.88E-07	0.3%	1	O
IE-FRI-A105-JY_S5	MCB Panel QMCB B1 Fire - MAIN STEAM AND FW	3.14E-06	5.94E-02	1.87E-07	0.3%	1	O
IE-FRI-1509_Q_RR	480 V AC SWGR 1NBL2 - No Target Damage	1.75E-02	1.06E-05	1.85E-07	0.3%	84	RR
IE-FRI-1095-JC_K1	U1 CSR A Term Cabinet 11601U3T19 Fire - Panel Only	6.18E-05	3.00E-03	1.85E-07	0.3%	11	G
IE-FRI-1079A-I9_B1	125 V DC Swgr 1BD1 Fire	2.46E-05	7.51E-03	1.85E-07	0.3%	3	G
IE-FRI-A105-JY_AT3	MCR Panel 1BCQSPB Fire - Section 01	9.96E-06	1.70E-02	1.70E-07	0.3%	1	O
IE-FRI-1174-JG_TR04_RR	Transient - Full Compartment	7.16E-05	2.34E-03	1.67E-07	0.3%	15	RR
IE-FRI-1030-C7_A_RR	Base Scenario	1.08E-02	1.51E-05	1.63E-07	0.3%	71	RR
IE-FRI-1080-IS_B1	480 V AC MCC 1NBR Fire	9.56E-05	1.67E-03	1.60E-07	0.3%	1	O
IE-FRI-A105-JY_AW3	MCR Panel 1ACQSPA Fire - Section 01	9.96E-06	1.58E-02	1.57E-07	0.3%	1	O
IE-FRI-1039C-CU_TR01	Bounding Transient	4.02E-05	3.86E-03	1.55E-07	0.3%	1	O
IE-FRI-1140C-S1_L2	Elevation 185 - South - SG4 QUADRANT	7.74E-04	1.92E-04	1.49E-07	0.2%	1	O

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-A040-BX_A_RR	Base Scenario	5.62E-03	2.62E-05	1.47E-07	0.2%	4	RR
IE-FRI-1025-BT_A_RR	Base Scenario	4.45E-03	3.28E-05	1.46E-07	0.2%	21	RR
IE-FRI-AHVSWYD_C	Relay Panel Fire Results in Loss of NXRB	1.59E-04	9.15E-04	1.46E-07	0.2%	1	O
IE-FRI-AHVSWYD_B	Relay Panel Fire Results in Loss of NXRA	1.59E-04	9.14E-04	1.45E-07	0.2%	1	O
IE-FRI-A105-JY_Q6	MCB Panel QMCB A2 Fire - RHR AND LETDOWN	2.09E-06	6.89E-02	1.44E-07	0.2%	1	O
IE-FRI-1085-JF_TR01	Transient at Middle of Corridor	3.68E-06	3.91E-02	1.44E-07	0.2%	1	O
IE-FRI-A105-JY_ABN4[3]	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	1.40E-07	1.00E+00	1.40E-07	0.2%	12	G
IE-FRI-1512_C0_RR	Swgr 1NB03 Fire - HEAF	1.47E-04	9.34E-04	1.37E-07	0.2%	7	RR
IE-FRI-1094-KQ_J1	U1 Isolating Auxiliary Relay Cabinet 1CCPAR2 Fire	5.31E-05	2.50E-03	1.33E-07	0.2%	1	O
IE-FRI-1092-J9_D0	Cabinet 1BCPAR9 Fire	2.32E-05	5.49E-03	1.28E-07	0.2%	1	O
IE-FRI-2133A-KK_A_RR	Base Scenario	1.22E-02	1.05E-05	1.28E-07	0.2%	168	RR
IE-FRI-A105-JY_AI	MCR Panel 11604Q5PS3 Fire	3.98E-05	3.20E-03	1.28E-07	0.2%	1	O
IE-FRI-TB1_A	Multi Compartment Scenario	1.92E-06	6.48E-02	1.24E-07	0.2%	1	O
IE-FRI-1120-KH_J4	U1 CSR B Term Cabinet 1NCPT18 Fire - Up to Tray 3	1.97E-05	6.08E-03	1.20E-07	0.2%	4	G
IE-FRI-1512_C2	Swgr 1NB03 Fire	1.31E-04	9.14E-04	1.19E-07	0.2%	1	O
IE-FRI-1120-KH_G1	U1 CSR B Term Cabinet 1BCPT12 Fire - Panel Only	2.00E-05	5.87E-03	1.17E-07	0.2%	1	O
IE-FRI-1095-JC_E7	U1 CSR A Term Cabinet 11601U3T07 Fire - Full ZOI	2.55E-06	4.52E-02	1.15E-07	0.2%	4	G
IE-FRI-1011B-A1_TR01	TRANSIENT IN THE CHASE	4.26E-06	2.65E-02	1.13E-07	0.2%	1	O
IE-FRI-1078A-IL_C1	Battery Charger 1AD1CA Fire	1.34E-05	8.18E-03	1.10E-07	0.2%	3	G
IE-FRI-1120-KH_H1	U1 CSR B Term Cabinet 1BCPT14 Fire - Panel Only	2.00E-05	5.32E-03	1.06E-07	0.2%	1	O
IE-FRI-YARD_TR10	Pull Box 2NCPXRA	2.58E-05	4.09E-03	1.06E-07	0.2%	1	O
IE-FRI-1140A-S1_E	Elevation 185 - North	9.54E-06	1.06E-02	1.01E-07	0.2%	2	G
IE-FRI-1603-KD_E_RR	125 V DC Panel A-1806-Q3-TS2/3/C Fire	9.65E-03	1.04E-05	1.01E-07	0.2%	94	RR
IE-FRI-1153-IQ_TR01	Bounding Transient	5.77E-05	1.65E-03	9.54E-08	0.2%	1	O
IE-FRI-1095-JC_F1_RR	U1 CSR A Term Cabinet 11601U3T09 Fire - Panel Only	3.77E-04	2.45E-04	9.24E-08	0.2%	16	RR

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1077A-IJ_B1	125 V DC Swgr 1CD1 Fire	2.16E-05	4.01E-03	8.69E-08	0.1%	3	G
IE-FRI-1080-IS_G2	1000 kVA Transformer 1-1805-S3-B10X Fire - Full ZOI	5.21E-05	1.65E-03	8.60E-08	0.1%	6	G
IE-FRI-1031-C6_A_RR	Base Scenario	2.55E-03	3.24E-05	8.25E-08	0.1%	16	RR
IE-FRI-1097-JJ_TR03	Transient - Small South Wall	5.06E-06	1.62E-02	8.19E-08	0.1%	1	O
IE-FRI-2085-NB_TR04_RR	TRANSIENT 2 AT NORTH WALL	7.82E-05	9.27E-04	7.24E-08	0.1%	12	RR
IE-FRI-1120-KH_L5	U1 CSR B Term Cabinet 1NCPT22 Fire - Full ZOI	4.94E-06	1.46E-02	7.23E-08	0.1%	2	G
IE-FRI-1094-KQ_C1	U1 Isolating Auxiliary Relay Cabinet 1NCPAR6 Fire	7.96E-05	8.97E-04	7.14E-08	0.1%	1	O
IE-FRI-A105-JY_AT1	MCR Panel 1BCQSPB Fire - Input Section	9.96E-06	6.73E-03	6.70E-08	0.1%	1	O
IE-FRI-A105-JY_AW1	MCR Panel 1ACQSPA Fire - Input Section	9.96E-06	6.70E-03	6.68E-08	0.1%	1	O
IE-FRI-1095-JC_G5	U1 CSR A Term Cabinet 11601U3T11 Fire - Up to Tray	3.86E-06	1.72E-02	6.63E-08	0.1%	2	G
IE-FRI-AHVSWYD_TR01	Bounding Pull Box Transient Fire	7.11E-05	9.25E-04	6.58E-08	0.1%	1	O
IE-FRI-1076-IC_D	125 V DC Panel 1ND32 Fire	3.98E-05	1.65E-03	6.56E-08	0.1%	1	O
IE-FRI-1076-IC_I	125 V DC Panel 1ND31 Fire	3.98E-05	1.65E-03	6.56E-08	0.1%	1	O
IE-FRI-1175-JI_TR01	Bounding Transient	3.65E-05	1.79E-03	6.52E-08	0.1%	1	O
IE-FRI-1095-JC_G7	U1 CSR A Term Cabinet 11601U3T11 Fire - Full ZOI	1.27E-06	5.03E-02	6.40E-08	0.1%	2	G
IE-FRI-ALVSWYD_TR01	Bounding Pull Box Transient Fire	1.55E-05	4.09E-03	6.34E-08	0.1%	1	O
IE-FRI-1156-V2_B	AFW Train A Pump Motor Fire	9.63E-05	6.55E-04	6.31E-08	0.1%	1	O
IE-FRI-1095-JC_N3_RR	U1 CSR A Term Cabinet 11601U3T25 Fire - Up to Tray	1.39E-04	4.53E-04	6.28E-08	0.1%	20	RR
IE-FRI-1121-KG_D1_RR	U1 Isolating Auxiliary Relay Cabinet 1NCPAR4 Fire	3.59E-04	1.74E-04	6.26E-08	0.1%	5	RR
IE-FRI-1120-KH_F3	U1 CSR B Term Cabinet 1BCPT10 Fire - Up to Tray 2	7.01E-06	8.77E-03	6.14E-08	0.1%	1	O
IE-FRI-1073-I7_TR03RR	TRANSIENT MIDWAY NORTH WALL	9.50E-06	6.46E-03	6.14E-08	0.1%	6	RR
IE-FRI-1083-IG_TR01	Bounding Transient	3.65E-05	1.66E-03	6.05E-08	0.1%	1	O
IE-FRI-A105-JY_R0	MCB Panel QMCB C Fire - Full Panel	2.09E-06	2.81E-02	5.87E-08	0.1%	1	O
IE-FRI-1095-JC_B5	U1 CSR A Term Cabinet 11601U3T01 Fire - Up to Tray	7.72E-06	7.36E-03	5.68E-08	0.1%	4	G
IE-FRI-1512_D1	Transformer 11805S3B03X Fire - Damage	6.25E-05	9.05E-04	5.66E-08	0.1%	1	O

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1056B-IH_TR01RR	Transient - Full Compartment	6.11E-05	9.24E-04	5.64E-08	0.1%	20	RR
IE-FRI-A105-JY_AT2	MCR Panel 1BCQSPB Fire - Logic Section	9.96E-06	5.31E-03	5.29E-08	0.1%	1	O
IE-FRI-A105-JY_AW2	MCR Panel 1ACQSPA Fire - Logic Section	9.96E-06	5.26E-03	5.24E-08	0.1%	1	O
IE-FRI-1512_K0	13.8 kVAC Swgr - 1NAA HEAF	2.97E-05	1.63E-03	4.84E-08	0.1%	1	O
IE-FRI-1512_B0	Swgr 1NAB Fire - HEAF	2.97E-05	1.63E-03	4.84E-08	0.1%	1	O
IE-FRI-1173-JH_TR02	Transient - Small East Center	5.06E-06	9.01E-03	4.56E-08	0.1%	1	O
IE-FRI-1151-IQ_TR03_RR	Transient - Full Compartment	4.18E-06	1.00E-02	4.19E-08	0.1%	2	RR
IE-FRI-1120-KH_J5	U1 CSR B Term Cabinet 1NCPT18 Fire - Full ZOI	2.47E-06	1.66E-02	4.10E-08	0.1%	1	O
IE-FRI-1026A-C7_A_RR	Base Scenario	6.35E-04	6.00E-05	3.81E-08	0.1%	18	RR
IE-FRI-1155-V1_TR01_RR	Transient - Full Compartment	7.13E-05	5.08E-04	3.62E-08	0.1%	23	RR
IE-FRI-1095-JC_D5	U1 CSR A Term Cabinet 11601U3T05 Fire - Up to Tray	3.86E-06	8.94E-03	3.45E-08	0.1%	2	G
IE-FRI-1176-K1_TR04	Transient - Small South Wall	5.06E-06	6.45E-03	3.26E-08	0.1%	1	O
IE-FRI-1059-JR_A_RR	Base Scenario	4.26E-03	7.40E-06	3.15E-08	0.1%	33	RR
IE-FRI-1140C-S1_C_RR	Elevation 197 - North	2.78E-03	1.11E-05	3.10E-08	0.1%	2	RR
IE-FRI-1176-K1_TR07	Transient near 1BE31CTYAER2	2.55E-06	1.20E-02	3.06E-08	0.0%	1	O
IE-FRI-1093-JA_TR02	Transient at Train B Shutdown Panel Room Door	8.35E-07	3.65E-02	3.05E-08	0.0%	1	O
IE-FRI-1097-JJ_TR04	Transient - Full Compartment	5.33E-07	5.71E-02	3.04E-08	0.0%	1	O
IE-FRI-1120-KH_K4	U1 CSR B Term Cabinet 1BCPT20 Fire - Up to Tray 3	2.83E-06	1.01E-02	2.85E-08	0.0%	1	O
IE-FRI-1014D-B9_A_RR	Base Scenario	8.33E-04	3.41E-05	2.84E-08	0.0%	12	RR
IE-FRI-2136-LP_A_RR	Base Scenario	3.35E-03	7.39E-06	2.48E-08	0.0%	124	RR
IE-FRI-1093-JA_TR03	Transient at Southeast Corner of Chase	8.35E-07	2.54E-02	2.12E-08	0.0%	1	O
IE-FRI-1062-JM_TR09	Transient at West Wall	1.53E-06	1.22E-02	1.87E-08	0.0%	1	O
IE-FRI-1140A-S1_I_RR	Elevation 197 - South	1.10E-03	1.65E-05	1.81E-08	0.0%	12	RR
IE-FRI-1121-KG_TR01	Transient - Full Compartment	4.97E-07	3.62E-02	1.80E-08	0.0%	1	O
IE-FRI-1086-KB_A_RR	Base Scenario	4.75E-04	3.74E-05	1.77E-08	0.0%	14	RR

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-A105-JY_U3A	MCB Panel QEAB 1A Fire - SECT. 3 - FIRE SPREAD	6.17E-08	2.65E-01	1.63E-08	0.0%	2	G
IE-FRI-1092-J9_C3	4.16 kV AC Swgr 1BA03	4.26E-07	3.67E-02	1.56E-08	0.0%	3	G
IE-FRI-1091-J8_B3	4.16 kV AC Swgr 1AA02	4.10E-07	3.03E-02	1.24E-08	0.0%	2	G
IE-FRI-1103-J8_TR01	Transient - Full Compartment	2.95E-07	3.70E-02	1.09E-08	0.0%	1	O
IE-FRI-1152-IN_TR01_RR	Transient - Below 6 Feet	1.41E-03	7.23E-06	1.02E-08	0.0%	149	RR
IE-FRI-1133B-KK_D2	U1 PSMS DPU B 1-1625-D5-006B Fire	4.24E-08	2.32E-01	9.85E-09	0.0%	5	G
IE-FRI-2095-N8_JB1	Junction Box	1.32E-03	7.21E-06	9.48E-09	0.0%	73	G
IE-FRI-1152-IN_TR02	Transient - Full Compartment	5.04E-07	1.53E-02	7.70E-09	0.0%	1	O
IE-FRI-1179-KV_B1_RR	TSC Inverter A-1807-Y3-TSCI7 Fire	1.13E-04	6.81E-05	7.68E-09	0.0%	8	G
IE-FRI-1077A-IJ_TR01RR	Transient - Full Compartment	1.94E-06	3.89E-03	7.53E-09	0.0%	6	RR
IE-FRI-YARD_TR01	LO Storage Tanks	7.24E-09	1.00E+00	7.24E-09	0.0%	2	G
IE-FRI-1066-IA_TR02	Transient - Full Compartment	1.86E-07	3.77E-02	7.04E-09	0.0%	1	O
IE-FRI-A105-JY_AT0	MCR Panel 1BCQSPB Fire - Full Panel	2.79E-07	2.44E-02	6.81E-09	0.0%	1	O
IE-FRI-A105-JY_AW0	MCR Panel 1ACQSPA Fire - Full Panel	2.79E-07	2.32E-02	6.46E-09	0.0%	1	O
IE-FRI-1144-T6_JB3_RR	Junction Box 1BWJB4867	8.62E-05	6.98E-05	6.01E-09	0.0%	25	RR
IE-FRI-1120-KH_TR09	TRANSIENT AT RISER ROW 5	3.58E-07	1.66E-02	5.93E-09	0.0%	1	O
IE-FRI-YARD_TR05	Pull Box 1NE7BBKEM02	8.00E-04	7.30E-06	5.84E-09	0.0%	31	G
IE-FRI-1092-J9_D1_RR	Cabinet 1BCPAR9 Fire - No Target Damage	4.88E-04	1.12E-05	5.48E-09	0.0%	32	RR
IE-FRI-1071-IF_G1_RR	480 V AC Switchgear 1BB07 Fire	1.15E-05	4.46E-04	5.11E-09	0.0%	4	RR
IE-FRI-2098-N9_JB1_RR	Junction Box	5.96E-04	7.04E-06	4.19E-09	0.0%	126	RR
IE-FRI-1091-J8_E2	Plant Safety Monitoring System PSMS Cabinet RPUA1	1.16E-07	2.94E-02	3.41E-09	0.0%	3	G
IE-FRI-1079A-I9_TR01	Transient - Full Compartment	4.63E-07	7.20E-03	3.33E-09	0.0%	1	O
IE-FRI-1056A-IM_TR01RR	Transient - Full Compartment	4.44E-06	6.93E-04	3.08E-09	0.0%	7	RR
IE-FRI-1113-JZ_TR03_RR	Transient - Full Compartment	6.35E-06	4.43E-04	2.81E-09	0.0%	18	RR
IE-FRI-1094-KQ_TR03	Transient - Full Compartment	3.65E-08	6.67E-02	2.43E-09	0.0%	1	O

Table 18-1 L3-FPRA 210 Fire Scenarios

Fire Scenario	Description	Initiating Event (/rcy)	CCDP	CDF (/rcy)[1]	% Cont.	No. of RP-FPRA Sequences	Grouped[2]
IE-FRI-1176-K1_TR06	Transient - Full Compartment	3.65E-08	5.35E-02	1.95E-09	0.0%	1	O
IE-FRI-A105-JY_U5A	MCB Panel QEAB 1A Fire - SECT. 5 - FIRE SPREAD	3.08E-08	5.89E-02	1.82E-09	0.0%	1	O
IE-FRI-1120-KH_TR04	TRANSIENT AT RISER ROW 1 - SOUTH OF CABINETS	3.58E-07	3.89E-03	1.39E-09	0.0%	1	O
IE-FRI-1017-AW_TR02_RR	Transient - Full Compartment	1.56E-07	6.82E-03	1.06E-09	0.0%	4	RR
IE-FRI-1503_TR01_RR	BOUNDING TRANSIENT	1.41E-04	6.67E-06	9.42E-10	0.0%	3	RR
IE-FRI-1078A-IL_TR01	Transient - Full Compartment	1.11E-07	7.08E-03	7.89E-10	0.0%	1	O
IE-FRI-1300A-X1_A_RR	Base Scenario	7.70E-05	5.98E-06	4.60E-10	0.0%	14	RR
IE-FRI-1098-JD_TR01	Bounding Transient	3.65E-08	1.02E-02	3.73E-10	0.0%	1	O
IE-FRI-1149-DO_TR02_RR	Transient - Full Compartment	5.87E-07	4.73E-04	2.77E-10	0.0%	14	RR
IE-FRI-A105-JY_U7A	MCB Panel QEAB 1A Fire - SECT. 7 - FIRE SPREAD	3.08E-08	3.59E-03	1.11E-10	0.0%	1	O

Note 1. /rcy = per reactor critical year
Note 2. Grouping: O = one-to-one fire sequence to fire scenario; G = group of fire sequences to fire scenario based on CCDP; RR = group of fire sequences to fire scenarios based on the residual fire sequences (i.e., those identified as not risk dominant).
Note 3. The severity factor for HVAC in normal mode has been reduced by the HEP of 0.1 to account for operator failure to place the HVAC in purge mode, as discussed in Section 18.4.5.5.

The final results from the L3-FPRA were compared to the RP-FPRA results to make sure the mapping process was reasonable. The first check was the overall difference between the two results, which are listed in Table 18-2. The overall CDF from the L3-FPRA is a factor of 1.5 (ratio) higher than the RP-FPRA CDF. A second check involved closer scrutiny of the contributions of the various mapping categories. According to Table 18-2, the fire sequences mapped on a one-to-one basis have relatively the same overall percentage contribution to both studies (48 percent). However, because of the higher percentage contribution of the residual fire scenarios to the L3-FPRA, the fire scenarios that are grouped based on CCDP have a corresponding lower overall percent contribution. The residual fire sequences contribute approximately 8 percent to the overall fire CDF in the RP-FPRA, while contributing approximately 14 percent to the overall fire CDF in the L3-FPRA. A more detailed discussion of the differences between the two models is provided in Section 18.4.6.

Table 18-2 Comparison of Percent Contributions of Mapping Process

	RP-FPRA			L3-FPRA		
Scenario Type	**No. of fire sequences**	**CDF (/rcy)**	**% of Total CDF**	**No. of fire scenarios**	**CDF (/rcy)**	**% of Total CDF**
One-to-One	102	1.99E-05	47.8%	102	2.96E-05	48.3%
Group	464	1.85E-05	44.5%	60	2.33E-05	38.0%
Residual	1915	3.16E-06	7.6%	48	8.43E-06	13.7%
Total	2481	4.16E-05	100.0%	210	6.14E-05	100.0%

18.4.3 Dominant Accident Sequences and Dominant Cut Sets

18.4.3.1 Dominant Accident Sequences

The L3-FPRA encompasses 210 fire scenarios, each fire scenario is comprised of 602 event tree accident sequences, for a total of 126,420 accident sequences. The top 10 accident sequences contribute 18.5 percent to the total fire CDF. The top 50 accident sequences, which contribute 44.6 percent to the total fire CDF, are shown in Table 18-3.

The top five accident sequences, which account for 12.3 percent of the CDF, are described below. The description identifies the various system successes and failures associated with each sequence, along with a high-level overview of whether the system failures are due to the fire or due to random causes.

1. Fire occurs in Main Control Room Panel – AMSAC 11626Q5AMS (FRI-A105-JY_AX) – The fire causes reactor trip. RCP seal cooling is successful along with no stuck open relief valves; therefore, there is no LOCA situation. The fire causes direct failure of the motor-driven pumps (MDPs) for both the main feedwater (MFW) and auxiliary feedwater (AFW) systems, as well as the steam admission valve for the AFW turbine-driven pump (TDP). This leads to complete failure of secondary-side cooling, necessitating bleed and feed operation. However, bleed and feed cooling also fails to provide decay heat removal, leading to core damage. The sequence frequency is 2.3×10^{-6}/rcy, contributing 3.8 percent to total plant fire CDF.

2. Fire occurs in Main Control Board Panel QMCB A1 Fire - NSCW (FRI-A105-JY_P2) – The fire causes a reactor trip. The fire also causes the failure of the nuclear service cooling water (NSCW) system, which leads to complete failure of RCP seal cooling. Following failure of RCP seal cooling, operators fail to trip the RCPs or the seals fail, leading to an RCP seal LOCA. Secondary-side cooling is initiated and is successful in removing decay heat. However, due to the loss of NSCW, neither high-pressure injection (HPI) nor low pressure injection (LPI) are available to provide make-up to compensate for the RCP seal LOCA, leading to core damage. The sequence frequency is 2.1×10^{-6}/rcy, contributing 3.4 percent to total plant fire CDF.

3. Fire occurs in 4.16 kV AC Switchgear 1AA02 Cubicle 00 (FRI-1091-J8_B100) – The fire causes a LOOP and a reactor trip. The fire also causes the direct loss of 4.16 kV AC switchgear bus 1AA02 (train A), which leads to failure of all components relying on this bus. Given the fire causes a LOOP and fails one train of AC power, the failure of the other train of AC power (i.e., the onsite diesel generator) leads to a station blackout. There is no LOCA (both the relief valves are closed and there is no RCP seal leak), but secondary-side cooling using the AFW turbine-driven pump fails to provide cooling. Offsite power is not recovered within 1 hour, which eventually leads to core damage. The sequence frequency for this scenario is 1.1×10^{-6}/rcy, contributing 1.8 percent to total plant fire CDF.

4. Fire occurs in Nuclear Service Water Tunnel - NSCW (FRI-1146-VF_TR01_RR) – The fire causes a reactor trip. The fire also causes the failure of the NSCW system, which leads to complete failure of RCP seal cooling. Following failure of RCP seal cooling, operators fail to trip the RCPs or the seals fail, leading to an RCP seal LOCA. Secondary-side cooling is initiated and is successful in removing decay heat. However, due to the loss of NSCW, neither HPI nor LPI are available to provide make-up to compensate for the RCP seal LOCA, leading to core damage. The sequence frequency is 1.1×10^{-6}/rcy, contributing 1.7 percent to total plant fire CDF.

5. Fire occurs in the 4.16 kV AC switchgear room 1BA03 Cubicle 04 (FRI-1092-J9_C204) – The fire causes a LOOP and a reactor trip. The fire also causes the direct loss of 4.16 kV AC switchgear bus 1BA03 (train B), which leads to failure of all components relying on this bus. Given the fire causes a LOOP and fails one train of AC power, the failure of the other train of AC power leads to a station blackout. Secondary-side cooling is successful using the AFW turbine-driven pump and there is no LOCA (both the relief valves are closed and there is no RCP seal leak). However, offsite power is not recovered prior to depleting the turbine building batteries (2-hour lifetime), which eventually leads to core damage. The sequence frequency for this scenario is 9.7×10^{-7}/rcy, contributing 1.6 percent to total plant fire CDF.

Table 18-3 L3-FPRA Top 50 CDF Sequences

Event Tree Name	Sequence Number	CDF (/rcy)	% Total CDF	No. of Cut Sets	Key Functional Failures
1-FRI-A105-JY_AX	02-04-1	2.33E-06	3.80	343	1-FIRE-RTRIP, 1-FW, 1-FAB: Fire induced reactor trip, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-A105-JY_P2	02-11-08-1	2.07E-06	3.37	3	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-HPI, 1-LPI: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of high pressure injection, and failure of low pressure injection.
1-FRI-1091-J8_B100	10-10-22-1	1.12E-06	1.83	209	1-FIRE-LOSP, 1-EPS, 1-AFW-B, 1-OPR-01H: Fire induced loss of offsite power, failure of onsite emergency power, failure of auxiliary feedwater turbine-driven pump, and failure to recover offsite power within 1 hour.
1-FRI-1146-VF_TR01_RR	02-11-08-1	1.06E-06	1.73	248	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-HPI, 1-LPI: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of high pressure injection, and failure of low pressure injection.
1-FRI-1092-J9_C204	10-10-07-1	9.68E-07	1.58	410	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1091-J8_B104	09-15-1	8.71E-07	1.42	15	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1078A-IL_G_RR	02-04-1	8.23E-07	1.34	1755	1-FIRE-RTRIP, 1-FW, 1-FAB: Fire induced reactor trip, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-1094-KQ_B1	10-10-07-1	7.39E-07	1.20	170	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1075-I8_F01	02-11-03-1	7.04E-07	1.15	611	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-RHR, 1-LPR: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of residual heat removal, and failure of low pressure recirculation.
1-FRI-1092-J9_C104	10-10-22-1	6.83E-07	1.11	314	1-FIRE-LOSP, 1-EPS, 1-AFW-B, 1-OPR-01H: Fire induced loss of offsite power, failure of onsite emergency power, failure of auxiliary feedwater turbine-driven pump, and failure to recover offsite power within 1 hour.
1-FRI-1091-J8_B200	10-10-07-1	6.69E-07	1.09	140	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1140B-S1_B	04-03-05-1	6.58E-07	1.07	48	1-FIRE-ISINJ, 1-TSI, 1-CAD-ES12, 1-HPR: Fire induced inadvertent safety injection, failure to terminate safety injection, failure to depressurize, and high pressure recirculation.
1-FRI-1157A-V3_B1	09-15-1	6.39E-07	1.04	7	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.

Table 18-3 L3-FPRA Top 50 CDF Sequences

Event Tree Name	Sequence Number	CDF (/rcy)	% Total CDF	No. of Cut Sets	Key Functional Failures
1-FRI-2080-M9_H1	10-10-07-1	6.18E-07	1.01	1370	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1120-KH_C6	07-7-1	6.18E-07	1.01	71	1-FIRE-MLOCA, 1-HPI, 1-CAD-MLOCA: Fire induced medium loss of coolant accident, failure of high pressure injection, and failure to depressurize.
1-FRI-1121-KG_E1	10-10-07-1	5.99E-07	0.98	379	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1074-ID_E	09-15-1	5.86E-07	0.95	29	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1103-J8_B1	06-03-1	5.67E-07	0.92	198	1-FIRE-SLOCA, 1-RHR, 1-LPR: Fire induced small loss of coolant accident, failure of residual heat removal, and failure of low pressure recirculation.
1-FRI-1098-JD_B1	06-03-1	5.65E-07	0.92	152	1-FIRE-SLOCA, 1-RHR, 1-LPR: Fire induced small loss of coolant accident, failure of residual heat removal, and failure of low pressure recirculation.
1-FRI-1075-I8_E1	02-11-03-1	5.63E-07	0.92	568	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-RHR, 1-LPR: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of residual heat removal, and failure of low pressure recirculation.
1-FRI-A105-JY_S2	09-04-1	5.12E-07	0.83	5	1-FIRE-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1091-J8_C0	09-15-1	4.95E-07	0.81	7	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1091-J8_B212	02-10-1	4.89E-07	0.80	950	1-FIRE-RTRIP, 1-RCPSI, 1-FW, 1-FAB: Fire induced reactor trip, failure of RCP seal injection, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-1098-JD_B1	09-21-1	4.69E-07	0.76	667	1-FIRE-SSBI, 1-SGI-SSBI, 1-RCPSI-CCPS, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of RCP seal injection using just charging pumps, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1075-I8_C01	02-11-03-1	4.57E-07	0.74	189	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-RHR, 1-LPR: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of residual heat removal, and failure of low pressure recirculation.

Table 18-3 L3-FPRA Top 50 CDF Sequences

Event Tree Name	Sequence Number	CDF (/rcy)	% Total CDF	No. of Cut Sets	Key Functional Failures
1-FRI-1092-J9_C104	09-21-1	4.38E-07	0.71	263	1-FIRE-SSBI, 1-SGI-SSBI, 1-RCPSI-CCPS, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of RCP seal injection using just charging pumps, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-A105-JY_AR	09-04-1	3.59E-07	0.59	7	1-FIRE-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1098-JD_B1	06-08-1	3.52E-07	0.57	328	1-FIRE-SLOCA, 1-HPI, 1-LPI: Fire induced small loss of coolant accident, failure of high pressure injection, and failure of low pressure injection.
1-FRI-1091-J8_B104	02-10-1	3.44E-07	0.56	807	1-FIRE-RTRIP, 1-RCPSI, 1-FW, 1-FAB: Fire induced reactor trip, failure of RCP seal injection, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-1091-J8_B100	10-10-21-1	3.13E-07	0.51	330	1-FIRE-LOSP, 1-EPS, 1-AFW-B, 1-AFW-ACR, 1-FAB-ACR: Fire induced loss of offsite power, failure of onsite emergency power, failure of auxiliary feedwater turbine-driven pump early, successful offsite power recovery within 1 hour, failure of auxiliary feedwater and failure feed and bleed given offsite power recovery.
1-FRI-1092-J9_C204	10-10-06-1	3.09E-07	0.50	162	1-FIRE-LOSP, 1-EPS, 1-AFW-ACR, 1-FAB-ACR: Fire induced loss of offsite power, failure of onsite emergency power, successful offsite power recovery within 2 hours, failure of auxiliary feedwater and failure feed and bleed given offsite power recovery.
1-FRI-1080-IS_K2	04-06-1	3.07E-07	0.50	1303	1-FIRE-ISINJ, 1-AFW, 1-FAB: Fire induced inadvertent safety injection, failure of auxiliary feedwater, and failure of feed and bleed cooling operation.
1-FRI-1091-J8_E0	05-06-1	3.03E-07	0.49	219	1-FIRE-LOSINJ, 1-IEFT-LOSINJ, 1-FW, 1-FAB: Fire induced loss of safety injection, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-A105-JY_AX	09-04-1	2.96E-07	0.48	10	1-FIRE-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1133B-KK_H0	10-10-22-1	2.96E-07	0.48	61	1-FIRE-LOSP, 1-EPS, 1-AFW-B, 1-OPR-01H: Fire induced loss of offsite power, failure of onsite emergency power, failure of auxiliary feedwater turbine-driven pump, and failure to recover offsite power within 1 hour.
1-FRI-1121-KG_B1	09-04-1	2.95E-07	0.48	4	1-FIRE-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1506_JB1	02-04-1	2.94E-07	0.48	239	1-FIRE-RTRIP, 1-FW, 1-FAB: Fire induced reactor trip, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.

Table 18-3 L3-FPRA Top 50 CDF Sequences

Event Tree Name	Sequence Number	CDF (/rcy)	% Total CDF	No. of Cut Sets	Key Functional Failures
1-FRI-1507_B1	09-15-1	2.92E-07	0.48	8	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1091-J8_B200	10-10-06-1	2.91E-07	0.47	315	1-FIRE-LOSP, 1-EPS, 1-AFW-ACR, 1-FAB-ACR: Fire induced loss of offsite power, failure of onsite emergency power, successful offsite power recovery within 2 hours, failure of auxiliary feedwater and failure feed and bleed given offsite power recovery.
1-FRI-1092-J9_E0	09-15-1	2.89E-07	0.47	7	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-AHVSWYD_E	10-10-07-1	2.87E-07	0.47	918	1-FIRE-LOSP, 1-EPS, 1-OPR-02H: Fire induced loss of offsite power, failure of onsite emergency power, and failure to recover offsite power within 2 hours.
1-FRI-1103-J8_B1	06-08-1	2.82E-07	0.46	335	1-FIRE-SLOCA, 1-HPI, 1-LPI: Fire induced small loss of coolant accident, failure of high pressure injection, and failure of low pressure injection.
1-FRI-1092-J9_C113	09-15-1	2.81E-07	0.46	7	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1508_TR01	10-06-1	2.78E-07	0.45	409	1-FIRE-LOSP, 1-AFW, 1-FAB: Fire induced loss of offsite power, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-1160A-V8_C_RR	02-11-08-1	2.68E-07	0.44	176	1-FIRE-RTRIP, 1-RCPSI, 1-RCPSC, 1-HPI, 1-LPI: Fire induced reactor trip, failure of RCP seal injection, failure of RCP thermal barrier cooling, failure of high pressure injection, and failure of low pressure injection.
1-FRI-1506_B1	02-04-1	2.66E-07	0.43	86	1-FIRE-RTRIP, 1-FW, 1-FAB: Fire induced reactor trip, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
1-FRI-A105-JY_S3	09-15-1	2.56E-07	0.42	5	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.
1-FRI-2080-M9_H1	10-10-06-1	2.53E-07	0.41	756	1-FIRE-LOSP, 1-EPS, 1-AFW-ACR, 1-FAB-ACR: Fire induced loss of offsite power, failure of onsite emergency power, successful offsite power recovery within 2 hours, failure of auxiliary feedwater and failure feed and bleed given offsite power recovery.
1-FRI-1092-J9_C100	09-15-1	2.52E-07	0.41	7	1-FIRE-SSBI, 1-SGI-SSBI, 1-AFW, 1-FAB: Fire induced steam line break inside containment, failure of steam generator isolation, failure of auxiliary feedwater system, and failure of feed and bleed cooling operation.

Table 18-3 L3-FPRA Top 50 CDF Sequences

Event Tree Name	Sequence Number	CDF (/rcy)	% Total CDF	No. of Cut Sets	Key Functional Failures
1-FRI-1078A-IL_G_RR	02-10-1	2.52E-07	0.41	103	1-FIRE-RTRIP, 1-RCPSI, 1-FW, 1-FAB: Fire induced reactor trip, failure of RCP seal injection, failure of all feedwater (main and auxiliary), and failure of feed and bleed cooling operation.
/rcy = per reactor critical year					

18.4.3.2 Dominant Cut Sets

The L3-FPRA has 210 fire scenarios that were analyzed to obtain the overall fire CDF. These 210 fire scenarios produced 232,784 minimal cut sets, after all fire sequences were gathered into a single end state. The top 50 cut sets are shown in Table 18-4. These top 50 cut sets contribute 31 percent to the overall fire CDF.

The top five cut sets contribute 10.0 percent to the overall fire CDF. These cut sets are described below.

1. The initiating event is a fire in the Main Control Room Panel – AMSAC 11626Q5AMS (FRI-A105-JY_AX). This fire causes loss of secondary-side cooling. The operators fail to initiate bleed and feed cooling, leading to core damage. The frequency of this cut set is 2.3×10^{-6}/rcy and contributes 3.8 percent to the overall fire CDF.

2. The initiating event is a fire in the Main Control Board Panel QMCB A1 – NSCW (FRI-A105-JY_P2). This fire causes loss of the NSCW system, resulting in a complete loss of RCP seal cooling and requiring the operator to trip the RCP pumps prior to seal failure. The operators fail to trip the RCPs, which leads to an RCP seal LOCA. Since the NSCW system is failed due to the fire, there is no long-term make-up available, leading to core damage. The frequency of this cut set is 1.4×10^{-6}/rcy and contributes 2.4 percent to the overall fire CDF.

3. The initiating event is a fire in the Main Control Board Panel QMCB A1 – NSCW (FRI-A105-JY_P2). This fire causes loss of the NSCW system, resulting in a complete loss of RCP seal cooling and requiring the operator to trip the RCP pumps prior to seal failure. The operators successfully trip the RCPs, but an RCP stage 2 seal fails that leads to an RCP seal LOCA. Since the NSCW system is failed due to the fire, there is no long-term make-up available, leading to core damage. The frequency of this cut set is 8.8×10^{-7}/rcy and contributes 1.4 percent to the overall fire CDF.

4. The initiating event is a fire in the 4.16 kV AC Switchgear Room 1AA02 Cubicle 04 (FRI-1091-J8_B104). This fire causes failure of the “A” train 4.16 kV AC bus; therefore, the “A” train of mitigating systems is failed. The fire also causes a spurious operation of the AFW TDP steam inlet valve, leading to starting of the AFW TDP, which causes an overfill of the steam generator and steam line break. Operators fail to control the AFW flow, leading to loss of secondary-side cooling. Operators also fail to initiate bleed and feed operation, which leads to core damage. The frequency of this cut set is 8.7×10^{-7}/rcy and contributes 1.4 percent to the overall fire CDF.

5. The initiating event is an AFW train C TDP fire (FRI-1157A-V3_B1). This fire causes spurious operation of the AFW TDP steam inlet valve, leading to starting of the AFW TDP and overfilling of the steam generators leading to failure of the AFW system. Operators also fail to initiate bleed and feed operation, which leads to core damage. The frequency of this cut set is 6.4×10^{-7}/rcy and contributes 1.0 percent to the overall fire CDF.

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
1	2.31E-06	5.80E-02	3.76%			
				1-IE-FRI-A105-JY_AX	3.98E-05	FIRE - MCR Panel - AMSAC 11626Q5AMS Fire
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
				1-OAF_MFW------H-FIRE	1.00E+00	OPERATORS FAIL TO ESTABLISH MFW TO SGs - FIRE
2	1.45E-06	3.28E-01	2.36%			
				1-IE-FRI-A105-JY_P2	4.41E-06	FIRE - MCB Panel QMCB A1 Fire - NSCW
				/1-OEP-VCF-LP-CLOPT	9.95E-01	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
				1-RCS-XHE-XM-TRIP-FIRE	3.30E-01	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)
3	8.81E-07	2.00E-01	1.44%			
				1-IE-FRI-A105-JY_P2	4.41E-06	FIRE - MCB Panel QMCB A1 Fire - NSCW
				1-RCS-MDP-LK-BP2	2.00E-01	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS
4	8.68E-07	2.48E-03	1.41%			
				1-IE-FRI-1091-J8_B104	3.50E-04	FIRE - 4.16 kV AC Swgr 1AA02 CUB 04 Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
5	6.37E-07	3.25E-02	1.04%			
				1-IE-FRI-1157A-V3_B1	1.96E-05	FIRE - AFW Train C Turbine Driven Pump Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
6	5.80E-07	2.91E-03	0.94%			
				1-IE-FRI-1074-ID_E	1.99E-04	FIRE - 125 V DC MCC 1CD1M Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OAISOLSTMTDAFW-FIRE	2.70E-02	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE
7	5.17E-07	5.35E-03	0.84%			
				1-IE-FRI-1092-J9_C204	9.67E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Damage - No Target Damage
				1-ACP-CRB-CC-AA0205__	5.35E-03	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
8	5.11E-07	5.80E-02	0.83%			
				1-IE-FRI-A105-JY_S2	8.81E-06	FIRE - MCB Panel QMCB B1 Fire - FW PT
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
9	4.94E-07	2.48E-03	0.80%			
				1-IE-FRI-1091-J8_C0	1.99E-04	FIRE - Train A Safety Features Sequencer Cabinet 1ACPSQ1
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
10	4.78E-07	8.81E-04	0.78%			
				1-IE-FRI-1140B-S1_B	5.43E-04	FIRE - Elevation 171 - North
				1-ACP-INV-MA-AD1I1___	8.81E-04	INVERTER 1AD1I1 IN MAINTENANCE
11	4.26E-07	5.35E-03	0.69%			
				1-IE-FRI-1094-KQ_B1	7.96E-05	FIRE - U1 Isolating Auxiliary Relay Cabinet 1ACPAR6 Fire
				1-ACP-CRB-CC-BA0301__	5.35E-03	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
12	4.21E-07	3.50E-04	0.69%			
				1-IE-FRI-2080-M9_H1	1.20E-03	FIRE - 480 V AC MCC 2NBR Fire
				1-ACP-CRB-CF-A205301	3.50E-04	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
13	4.16E-07	5.35E-03	0.68%			
				1-IE-FRI-1091-J8_B100	7.77E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire
				1-ACP-CRB-CC-BA0301__	5.35E-03	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
14	4.02E-07	1.68E-03	0.65%			
				1-IE-FRI-1098-JD_B1	2.39E-04	FIRE - Train B Shutdown Panel 1-1605-P5-SDB Fire - No Spread
				1-LPI-MDP-MA-RHRA____	3.00E-03	RHR PUMP A IN MAINTENANCE
				1-RCS-PO-CO-RV0456A_56	5.60E-01	PORV PV0456A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
15	4.02E-07	1.68E-03	0.65%			
				1-IE-FRI-1103-J8_B1	2.39E-04	FIRE - Train A Shutdown Panel 1-1605-P5-SDA Fire - No Spread
				1-LPI-MDP-MA-RHRB____	3.00E-03	RHR PUMP B IN MAINTENANCE
				1-RCS-PO-CO-PV0455A_56	5.60E-01	PORV PV0455A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
16	3.87E-07	5.35E-03	0.63%			
				1-IE-FRI-1091-J8_B200	7.22E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire - No Target Damage
				1-ACP-CRB-CC-BA0301__	5.35E-03	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
17	3.53E-07	4.43E-03	0.57%			
				1-IE-FRI-A105-JY_AR	7.96E-05	FIRE - MCR Panel 1NCQARB Fire
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
18	3.19E-07	5.35E-03	0.52%			
				1-IE-FRI-1121-KG_E1	5.97E-05	FIRE - U1 Isolating Auxiliary Relay Cabinet 1BCPAR7 Fire
				1-ACP-CRB-CC-AA0205__	5.35E-03	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
19	3.18E-07	7.06E-03	0.52%			
				1-IE-FRI-1092-J9_C104	4.51E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Fire

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
				1-EPS-DGN-MA-G4001___	1.26E-02	DG1A IN MAINTENANCE
				1-MSS-ADV-CO-VPV3020_56	5.60E-01	SG3 ARV PV-3020 SPURIOUSLY OPENS - (FIRE 0.56)
20	2.95E-07	2.48E-03	0.48%			
				1-IE-FRI-1121-KG_B1	1.19E-04	FIRE - U1 Isolating Auxiliary Relay Cabinet 1BCPAR3 Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
21	2.95E-07	3.79E-03	0.48%			
				1-IE-FRI-1091-J8_B100	7.77E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire
				1-EPS-DGN-FR-G4002___	3.30E-02	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
				1-OA-ALIGNPW-01HR-FIRE	1.15E-01	OPERATORS FAIL TO ALIGN THE ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 1 HR AFTER SBO - FIRE
22	2.88E-07	2.48E-03	0.47%			
				1-IE-FRI-1092-J9_E0	1.16E-04	FIRE - Train B Safety Features Sequencer Cabinet 1-1821-U
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
23	2.80E-07	2.48E-03	0.46%			
				1-IE-FRI-1092-J9_C113	1.13E-04	FIRE - 4.16 kV AC Swgr 1BA03 Cub 13 Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
24	2.64E-07	5.80E-02	0.43%			
				1-IE-FRI-1506_B1	4.56E-06	FIRE - MAIN FEEDWATER PUMP FIRE - OIL FIRE
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
25	2.63E-07	2.72E-03	0.43%			
				1-IE-FRI-1092-J9_C204	9.67E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Damage - No Target Damage
				1-DCP-BAT-MA-AD1B____	2.72E-03	BATTERY 1AD1B IN MAINTENANCE
26	2.56E-07	5.80E-02	0.42%			
				1-IE-FRI-A105-JY_S3	4.41E-06	FIRE - MCB Panel QMCB B1 Fire - AFW

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
27	2.52E-07	2.48E-03	0.41%			
				1-IE-FRI-1092-J9_C100	1.01E-04	FIRE - 4.16 kV AC Swgr 1BA03 Cub 00 Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
28	2.41E-07	5.35E-03	0.39%			
				1-IE-FRI-1092-J9_C104	4.51E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Fire
				1-ACP-CRB-CC-AA0205__	5.35E-03	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
29	2.37E-07	2.48E-03	0.39%			
				1-IE-FRI-1074-ID_B1	9.56E-05	FIRE - 480 V AC MCC 1NBS Fire
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
30	2.32E-07	1.57E-01	0.38%			
				1-IE-FRI-1120-KH_C6	1.48E-06	FIRE - U1 CSR B Term Cabinet 1BCPT04 Fire - Suppression F
				1-HPI-MOV-OC-HV8806_28	2.80E-01	HV8806 SPURIOUSLY CLOSES AND ISOLATES RWST FROM SIP SUCTION HEADER - FIRE (0.28)
				1-RCS-PO-CO-RV0456A_56	5.60E-01	PORV PV0456A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
31	2.32E-07	1.57E-01	0.38%			
				1-IE-FRI-1120-KH_C6	1.48E-06	FIRE - U1 CSR B Term Cabinet 1BCPT04 Fire - Suppression F
				1-HPI-MOV-OC-HV8813_28	2.80E-01	SI PUMPS MINIFLOW LINE MOV HV8813 SPURIOUSLY CLOSES DUE TO FIRE (0.28)
				1-RCS-PO-CO-RV0456A_56	5.60E-01	PORV PV0456A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
32	2.32E-07	1.57E-01	0.38%			
				1-IE-FRI-1120-KH_C6	1.48E-06	FIRE - U1 CSR B Term Cabinet 1BCPT04 Fire - Suppression F
				1-HPI-MOV-OC-HV8438_28	2.80E-01	HPI-CCP INTERCONNECT VALVE HV8438 FAILS CLOSED DUE TO FIRE (PROB 0.28)
				1-RCS-PO-CO-RV0456A_56	5.60E-01	PORV PV0456A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
33	2.32E-07	2.91E-03	0.38%			
				1-IE-FRI-1095-JC_J3	7.95E-05	FIRE - U1 CSR A Term Cabinet 11601U3T17 Fire - Up to Tray
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OAISOLSTMTDAFW-FIRE	2.70E-02	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE
34	2.21E-07	2.21E-03	0.36%			
				1-IE-FRI-1506_JB1	1.00E-04	FIRE - Junction Box 1NQJB6012
				1-AFW-TDP-FR-P4001___	3.80E-02	TDAFWP (P4-001) FAILS TO RUN
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
35	2.17E-07	2.72E-03	0.35%			
				1-IE-FRI-1094-KQ_B1	7.96E-05	FIRE - U1 Isolating Auxiliary Relay Cabinet 1ACPAR6 Fire
				1-DCP-BAT-MA-BD1B____	2.72E-03	BATTERY 1BD1B IN MAINTENANCE
36	2.11E-07	2.72E-03	0.34%			
				1-IE-FRI-1091-J8_B100	7.77E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire
				1-DCP-BAT-MA-BD1B____	2.72E-03	BATTERY 1BD1B IN MAINTENANCE
37	1.98E-07	4.44E-01	0.32%			
				1-IE-FRI-1095-JC_B8	4.46E-07	FIRE - U1 CSR A Term Cabinet 11601U3T01 Fire - Suppression
				1-OAD_MLA------H-FIRE	4.44E-01	OPERATORS FAIL TO DEPRESSURIZE SECONDARY FOR LPI - MLO w HPI FAILED - FIRE
38	1.97E-07	2.72E-03	0.32%			
				1-IE-FRI-1091-J8_B200	7.22E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire - No Target Damage
				1-DCP-BAT-MA-BD1B____	2.72E-03	BATTERY 1BD1B IN MAINTENANCE
39	1.96E-07	3.50E-04	0.32%			
				1-IE-FRI-AHVSWYD_E	5.60E-04	FIRE - Main Control Panel Fire Results in Loss of Both Of
				1-ACP-CRB-CF-A205301	3.50E-04	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
40	1.96E-07	2.15E-04	0.32%			
				1-IE-FRI-1078A-IL_G_RR	9.10E-04	125 V DC Panel 1AD11 Fire
				1-ACP-BAC-MA-BA03____	2.15E-04	4.16KV BUS 1BA03 IN MAINTENANCE
41	1.95E-07	7.06E-05	0.32%			
				1-IE-FRI-1146-VF_TR01_RR	2.76E-03	FIRE - Bounding Transient
				1-ACP-BAC-MA-AA02____	2.15E-04	BUS 1AA02 IN MAINTENANCE
				/1-OEP-VCF-LP-CLOPT	9.95E-01	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
				1-RCS-XHE-XM-TRIP-FIRE	3.30E-01	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)
42	1.95E-07	7.06E-05	0.32%			
				1-IE-FRI-1146-VF_TR01_RR	2.76E-03	FIRE - Bounding Transient
				1-ACP-BAC-MA-AB15____	2.15E-04	480 V AC SWGR 1AB15 IN MAINTENANCE
				/1-OEP-VCF-LP-CLOPT	9.95E-01	CONSEQUENTIAL LOSS OF OFFSITE POWER - TRANSIENT
				1-RCS-XHE-XM-TRIP-FIRE	3.30E-01	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)
43	1.93E-07	2.48E-03	0.31%			
				1-IE-FRI-1091-J8_B100	7.77E-05	FIRE - 4.16 kV AC Swgr 1AA02 CUB 00 Fire

Table 18-4 L3-FPRA Top 50 CDF Cut Sets

	CDF (/rcy)	CCDP	% of Total	IE / Basic Events	Prob.	Description
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
				1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
44	1.88E-07	4.43E-03	0.31%			
				1-IE-FRI-1095-JC_G1	4.24E-05	FIRE - U1 CSR A Term Cabinet 11601U3T11 Fire - Panel Only
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
45	1.83E-07	1.89E-03	0.30%			
				1-IE-FRI-1092-J9_C204	9.67E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Damage - No Target Damage
				1-EPS-DGN-FR-G4001___	3.30E-02	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
				1-OA-ORS-------H-FIRE	5.73E-02	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO (FIRE RELATED)
46	1.82E-07	5.80E-02	0.30%			
				1-IE-FRI-A105-JY_S5	3.14E-06	FIRE - MCB Panel QMCB B1 Fire - MAIN STEAM AND FW
				1-OAB_TR-------H-FIRE	5.80E-02	OPERATORS FAIL TO FEED & BLEED -TRANSIENT - FIRE
47	1.80E-07	2.91E-03	0.29%			
				1-IE-FRI-1095-JC_K1	6.18E-05	FIRE - U1 CSR A Term Cabinet 11601U3T19 Fire - Panel Only
				1-AFW-MOV-CO-HV5106__56	5.60E-01	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OAISOLSTMTDAFW-FIRE	2.70E-02	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE
48	1.76E-07	4.43E-03	0.29%			
				1-IE-FRI-A105-JY_AV	3.98E-05	FIRE - MCR Panel 1ACQSTA Fire
				1-OAB_TR-------H-FIRE-MD	1.93E-01	OPERATORS FAIL TO FEED & BLEED -TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
				1-OACONTROL--AFW-FIRE	2.30E-02	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
49	1.71E-07	3.79E-03	0.28%			
				1-IE-FRI-1092-J9_C104	4.51E-05	FIRE - 4.16 kV AC Swgr 1BA03 Cub 04 Fire
				1-EPS-DGN-FR-G4001___	3.30E-02	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
				1-OA-ALIGNPW-01HR-FIRE	1.15E-01	OPERATORS FAIL TO ALIGN ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 1 HR AFTER SBO - FIRE
50	1.62E-07	2.72E-03	0.26%			
				1-IE-FRI-1121-KG_E1	5.97E-05	FIRE - U1 Isolating Auxiliary Relay Cabinet 1BCPAR7 Fire
				1-DCP-BAT-MA-AD1B____	2.72E-03	BATTERY 1AD1B IN MAINTENANCE

18.4.4 L3-FPRA Important Failure Events

This section provides information on the important human failure events (HFEs) and hardware components from the L3-FPRA model in Sections 18.4.4.1 and 18.4.4.2, respectively. Several importance measures are reported for each event. These importance measures are discussed below along with the equations used in their calculation.

Fussell-Vesely (FV)

The FV importance measure provides the percent contribution of the basic event to the overall CDF.

The FV is calculated using the following equation: $FV_i = F_i(x)/F(x)$;

where $F_i(x)$ is the calculated CDF of all cut sets that contain that particular basic event (BE_i) and $F(x)$ is the overall CDF (all plant cut sets).

Risk Increase Ratio (RIR)

The RIR importance measure provides information about how much of an increase in risk would be expected if the basic event were expected to fail every time it was demanded.

The RIR is calculated using the following equation: $RIR_i = F_i(1.0)/F(x)$;

where $F_i(1.0)$ is the calculated CDF of all cut sets with that particular basic event (BE_i) set to a probability of 1.0 (guaranteed to fail) and $F(x)$ is the overall CDF with that particular basic event set to its nominal (original) value.

Risk Reduction Ratio (RRR)

The RRR importance measure provides information about how much of a decrease in risk would be expected if the basic event were expected to be successful every time it was demanded.

The RRR is calculated using the following equation: $RRR_i = F(x)/F_i(0.0)$;

where $F_i(0.0)$ is the overall CDF with that particular basic event (BE_i) set to a probability of 0.0 (guaranteed to succeed) and $F(x)$ is the overall CDF with that particular basic event set to its nominal (original) value.

Birnbaum

The Birnbaum importance measure provides information about how much of a change (delta) would be expected if the basic event were guaranteed to fail and guaranteed to succeed.

The Birnbaum is calculated using the following equation: $Birnbaum = F_i(1.0) - F_i(0.0)$;

where $F_i(0.0)$ is the calculated CDF of all cut sets with that particular basic event (BE_i) set to a probability of 0.0 (guaranteed to succeed) and $F_i(1.0)$ is the calculated CDF of all cut sets with that particular basic event (BE_i) set to a probability of 1.0 (guaranteed to fail).

18.4.4.1 Important Human Failure Events

HFEs are major contributors to core damage in the L3-FPRA model. This is due to the limited mitigating systems available in many fire scenarios. The dominant HFEs are listed in Table 18-5, sorted by the FV importance measure. The table only lists those HFEs with a FV importance greater than 1.0×10^{-3} (0.1 percent contribution). Besides the various importance measures, the HEP is also provided in the table.

According to Table 18-5, the HFE with the highest FV importance (0.14) is operator (independent) failure to initiate bleed and feed cooling in the absence of secondary-side heat removal. This action often occurs in cut sets with the HFE for failure to control AFW, for which there is a moderate dependency. The dependent version of this action has the third highest FV importance (0.10).

The HFE with the second highest FV importance is operator failure to trip the RCPs given loss of seal cooling. Per the WOG-2000 RCP seal LOCA model, failure to trip the RCPs within 13 minutes after the loss of RCP seal cooling results in an RCP seal LOCA. This can have a significant impact on CDF, since under most fire scenarios, long-term cooling is unavailable because the fire causes a direct failure of either the auxiliary component cooling water system or the nuclear service cooling water system. Failure to trip the RCPs prior to a seal LOCA contributes approximately 11 percent to total fire CDF (FV = 0.11).

The HFE with the fourth highest FV importance is failure to control AFW given a fire causes the spurious operation of the system (e.g., spurious starting of the AFW pumps). Failure to control AFW flow under these conditions will lead to a loss of secondary-side heat removal and contributes approximately 9.9 percent to total fire CDF (FV = 0.099).

Table 18-5 Risk Important Operator Actions

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-OAB_TR-------H-FIRE	5.80E-02	1.36E-01	3.21E+00	1.16E+00	1.47E-04	OPERATORS FAIL TO FEED & BLEED - TRANSIENT - FIRE
1-RCS-XHE-XM-TRIP-FIRE	3.30E-01	1.12E-01	1.23E+00	1.13E+00	2.11E-05	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)
1-OAB_TR-------H-FIRE-MD	1.93E-01	1.04E-01	1.44E+00	1.12E+00	3.38E-05	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
1-OACONTROL--AFW-FIRE	2.30E-02	9.87E-02	5.19E+00	1.11E+00	2.68E-04	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE
1-OAISOLSTMTDAFW-FIRE-CD	1.00E+00	5.87E-02	1.00E+00	1.06E+00	3.67E-06	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)
1-OA-ALIGNPW-01HR-FIRE	1.15E-01	4.14E-02	1.32E+00	1.04E+00	2.25E-05	OPERATORS FAIL TO ALIGN THE ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 1 HR AFTER SBO - FIRE
1-OA-ORS-------H-FIRE	5.73E-02	4.13E-02	1.68E+00	1.04E+00	4.50E-05	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO (FIRE RELATED)
1-OAF_MFW------H-FIRE	1.00E+00	3.78E-02	1.00E+00	1.04E+00	2.36E-06	OPERATORS FAIL TO ESTABLISH MFW TO SGs - FIRE

Table 18-5 Risk Important Operator Actions

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-OA-NSCWFAN---H-FIRE	1.00E+00	3.45E-02	1.00E+00	1.04E+00	2.16E-06	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER) - FIRE
1-OAISOLSTMTDAFW-FIRE	2.70E-02	1.77E-02	1.64E+00	1.02E+00	4.09E-05	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE
1-OA-START-AFW-H-FIRE	1.24E-02	1.41E-02	2.13E+00	1.01E+00	7.12E-05	OPERATORS FAIL TO MANUALLY START AFW PUMPS IN MCR - FIRE
1-OAB_TR-------H-FIRE-HD	5.29E-01	1.29E-02	1.01E+00	1.01E+00	1.52E-06	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (HIGH DEPENDENCY)
1-OA-OBR-------H-FIRE	1.00E+00	1.07E-02	1.00E+00	1.01E+00	6.67E-07	OPERATORS FAIL TO ESTABLISH EMERGENCY BORATION
1-OA-ALIGNPW-02HR-FIRE	1.22E-02	8.21E-03	1.66E+00	1.01E+00	4.20E-05	OPERATORS FAIL TO ALIGN THE ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 2HR AFTER SBO - FIRE
1-OA-ESFAS-HE1-H-FIRE	1.86E-02	7.70E-03	1.41E+00	1.01E+00	2.59E-05	OPERATORS FAIL TO START EQUIP ON FAILURE OF ESFAS SIGNAL - FIRE
1-OAR_HPMSO----H-FIRE	1.20E-02	6.84E-03	1.56E+00	1.01E+00	3.56E-05	OPERATORS FAIL TO ESTABLISH HPR - RWST MSO - FIRE
1-OA-MISPAF5094H	1.00E-03	5.95E-03	6.94E+00	1.01E+00	3.71E-04	POST-TEST MISPOSITIONING OF MDAFWP B SUCTION MANUAL VALVE HV5094
1-RPS-XHE-XE-NSGNL-FIRE	2.30E-01	5.59E-03	1.02E+00	1.01E+00	1.52E-06	OPERATORS FAIL TO RESPOND WITH NO RPS SIGNAL PRESENT (FIRE RELATED)
1-OAR_LPSL-----H-FIRE	1.10E-03	5.51E-03	6.00E+00	1.01E+00	3.13E-04	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION -SLOCA
1-OAR_HPSLA----H-FIRE	6.00E-04	4.50E-03	8.49E+00	1.01E+00	4.68E-04	OPERATORS FAIL TO ESTABLISH HPR - SLOCA with CCUs AVAILABLE - FIRE
1-OA-ISOLETDOWNH-FIRE	1.90E-02	4.41E-03	1.23E+00	1.00E+00	1.45E-05	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE
1-OA-ALTAFW----H-FIRE	1.32E-03	4.09E-03	4.10E+00	1.00E+00	1.94E-04	OPERATORS FAIL TO PROVIDE ADDITIONAL WATER SOURCE FOR LONG TERM AFW - FIRE
1-OAD_MLA------H-FIRE	4.44E-01	3.17E-03	1.00E+00	1.00E+00	4.46E-07	OPERATORS FAIL TO DEPRESSURIZE SECONDARY FOR LPI - MLO w HPI FAILED - FIRE
1-OAC_NC-------H-FIRE	2.19E-03	3.07E-03	2.40E+00	1.00E+00	8.76E-05	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE
1-OA-MISPAF5095H	1.00E-03	2.72E-03	3.72E+00	1.00E+00	1.70E-04	POST-TEST MISPOSITIONING OF MDAFWP A SUCTION MANUAL HV5095
1-OA-OLP_SL----H-FIRE	1.23E-02	2.71E-03	1.22E+00	1.00E+00	1.38E-05	OPERATORS FAIL TO RESTART RHR PUMP FOR LPI SLOCA HPI FAILS DPI SUCCESS - FIRE
1-CAD-XHE-SAFESTBLE-FIRE	7.50E-04	2.37E-03	4.16E+00	1.00E+00	1.98E-04	OPERATORS FAIL TO DEPRESSURIZE SECONDARY (72HR SAFE/STABLE) (FIRE RELATED)
1-OAB_SI-------H-FIRE	2.35E-02	2.24E-03	1.09E+00	1.00E+00	5.96E-06	OPERATORS FAIL TO BLEED & FEED -SI - FIRE
1-OA-SAGD-CHG--H-FIRE	1.00E+00	2.23E-03	1.00E+00	1.00E+00	1.40E-07	OPERATORS FAIL TO ESTABLISH SAFETY GRADE CHARGING AFTER LOSINJ IE - FIRE
1-OAB-SBOACR---H-FIRE	1.00E+00	2.16E-03	1.00E+00	1.00E+00	1.35E-07	OPERATORS FAIL TO INITIATE FEED AND BLEED - SBO ACR

Table 18-5 Risk Important Operator Actions

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-OAR_LPSL-----H-FIRE-LD	5.10E-02	1.84E-03	1.03E+00	1.00E+00	2.25E-06	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION PER -SLOCA (LOW DEPENDENCY)
1-OAT----------H-FIRE-HD	5.00E-01	1.84E-03	1.00E+00	1.00E+00	2.30E-07	OPERATORS FAIL TO TERMINATE SI - FIRE (HIGH DEPENDENCY)
1-OAR_LTFB-TRA-H-FIRE	6.00E-04	1.72E-03	3.86E+00	1.00E+00	1.79E-04	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -TRANSIENT CCU AVAILABLE - FIRE
1-OAN_SL-------H-FIRE	1.10E-03	1.52E-03	2.38E+00	1.00E+00	8.64E-05	OPERATORS FAIL TO ESTABLISH NORMAL RHR -SLOCA - FIRE

18.4.4.2 Important Hardware Failure Events

Mitigating systems are affected by the different fire scenarios modeled in the L3-FPRA, due to individual equipment or trains being directly failed. Table 18-6 provides a list of the hardware components with a FV importance greater than 1.0×10^{-3} (i.e., greater than 0.1 percent contribution to total fire CDF) for all the fire scenarios grouped together (i.e., end state CD-FRI). The table is sorted by FV importance. Note, since the fire scenarios are dominated by the direct effects of the fire and human errors, random hardware failures are not typically as important as they are for internal event scenarios.

The hardware component failure with the highest FV importance is the RCP stage 2 seal failure given all seal cooling is lost. This component failure is in cut sets that contribute 9.2 percent to total fire CDF (FV = 0.092).

The hardware component failure with the next highest FV importance is spurious opening of the turbine-driven AFW pump steam inlet valve. The spurious opening of this valve will cause the turbine-driven AFW pump to start, which can lead to overfilling the steam generators and an induced steam line break. This component failure is in cut sets that contribute 9.1 percent to the overall fire CDF (FV = 0.091).

The two component failures with the next highest FV importance are diesel generators DG 4001 (1A) and DG-4002 (1B) failing to operate for the mission time. These two component failures contribute 5.0 percent and 4.7 percent, respectively, to total fire CDF (FV = 0.050 and FV = 0.47, respectively). The diesel generators have relatively high importance because multiple fire scenarios cause a LOOP, thereby requiring onsite emergency power to start and operate to provide essential AC power.

All other hardware component failures contribute less than 4 percent to total fire CDF (FV < 0.04).

Table 18-6 Risk Important Hardware Components

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-RCS-MDP-LK-BP2	2.00E-01	9.17E-02	1.37E+00	1.10E+00	2.87E-05	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS
1-AFW-MOV-CO-HV5106__56	5.60E-01	9.13E-02	1.07E+00	1.10E+00	1.02E-05	AFW MOV HV5106 SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
1-EPS-DGN-FR-G4001___	3.30E-02	5.04E-02	2.48E+00	1.05E+00	9.56E-05	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-EPS-DGN-FR-G4002___	3.30E-02	4.71E-02	2.38E+00	1.05E+00	8.92E-05	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
1-ACP-CRB-CC-AA0205__	5.35E-03	3.89E-02	8.23E+00	1.04E+00	4.54E-04	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
1-RCS-PO-CO-RV0456A_56	5.60E-01	3.89E-02	1.03E+00	1.04E+00	4.34E-06	PORV PV0456A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
1-ACP-CRB-CC-BA0301__	5.35E-03	3.85E-02	8.15E+00	1.04E+00	4.49E-04	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
1-EPS-DGN-MA-G4001___	1.26E-02	3.78E-02	3.96E+00	1.04E+00	1.87E-04	DG1A IN MAINTENANCE
1-LPI-MDP-MA-RHRB____	3.00E-03	3.60E-02	1.30E+01	1.04E+00	7.50E-04	RHR PUMP B IN MAINTENANCE
1-HPI-MOV-OC-HV8438_28	2.80E-01	3.31E-02	1.09E+00	1.03E+00	7.38E-06	HPI-CCP INTERCONNECT VALVE HV8438 FAILS CLOSED DUE TO FIRE (PROB 0.28)
1-EPS-DGN-MA-G4002___	1.26E-02	2.55E-02	3.00E+00	1.03E+00	1.27E-04	DG1B IN MAINTENANCE
1-AFW-MDP-CO-P4002___56	5.60E-01	2.09E-02	1.02E+00	1.02E+00	2.34E-06	SPURIOUS OPERATION OF MDAFWP B (PROB 0.56)
1-NSCWCT-SPRAY	9.04E-01	1.99E-02	1.00E+00	1.02E+00	1.37E-06	NSCW CTS IN SPRAY MODE (FRACTION OF TIME)
1-LPI-MDP-MA-RHRA____	3.00E-03	1.89E-02	7.28E+00	1.02E+00	3.94E-04	RHR PUMP A IN MAINTENANCE
1-DCP-BAT-MA-BD1B____	2.72E-03	1.86E-02	7.81E+00	1.02E+00	4.27E-04	BATTERY 1BD1B IN MAINTENANCE
1-DCP-BAT-MA-AD1B____	2.72E-03	1.84E-02	7.73E+00	1.02E+00	4.22E-04	BATTERY 1AD1B IN MAINTENANCE
1-ACP-BAC-MA-BA03____	2.15E-04	1.69E-02	7.92E+01	1.02E+00	4.89E-03	4.16KV BUS 1BA03 IN MAINTENANCE
1-AFW-MDP-MA-P4002___	3.00E-03	1.66E-02	6.50E+00	1.02E+00	3.45E-04	MDAFWP B (P4-002) UNAVAILABLE DUE TO T&M
1-ACP-BAC-MA-AA02____	2.15E-04	1.63E-02	7.64E+01	1.02E+00	4.71E-03	BUS 1AA02 IN MAINTENANCE
1-RCS-PO-CO-PV0455A_56	5.60E-01	1.57E-02	1.01E+00	1.02E+00	1.75E-06	PORV PV0455A SPURIOUSLY OPENS DUE TO FIRE (PROB 0.56)
1-MSS-ADV-CO-VPV3020_56	5.60E-01	1.56E-02	1.01E+00	1.02E+00	1.74E-06	SG3 ARV PV-3020 SPURIOUSLY OPENS - (FIRE 0.56)
1-ACP-INV-MA-AD1I1___	8.81E-04	1.49E-02	1.79E+01	1.02E+00	1.06E-03	INVERTER 1AD1I1 IN MAINTENANCE
1-AFW-TDP-FR-P4001___	3.80E-02	1.32E-02	1.34E+00	1.01E+00	2.18E-05	TDAFWP (P4-001) FAILS TO RUN
1-ACP-BAC-MA-AB15____	2.15E-04	1.25E-02	5.90E+01	1.01E+00	3.63E-03	480 V AC SWGR 1AB15 IN MAINTENANCE
1-LPI-MDP-FS-RHRB____	1.00E-03	1.23E-02	1.33E+01	1.01E+00	7.68E-04	RHR PUMP B RANDOMLY FAILS TO START
1-ACP-BAC-MA-BB16____	2.15E-04	1.23E-02	5.79E+01	1.01E+00	3.55E-03	480 V AC SWGR 1BB16 IN MAINTENANCE
1-EPS-DGN-FS-G4001___	2.94E-03	1.18E-02	4.99E+00	1.01E+00	2.50E-04	DG1A RANDOMLY FAILS TO START

Table 18-6 Risk Important Hardware Components

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-HPI-MOV-OC-HV8806_28	2.80E-01	1.15E-02	1.03E+00	1.01E+00	2.57E-06	HV8806 SPURIOUSLY CLOSES AND ISOLATES RWST FROM SIP SUCTION HEADER - FIRE (0.28)
1-HPI-MOV-OC-HV8813_28	2.80E-01	1.15E-02	1.03E+00	1.01E+00	2.57E-06	SI PUMPS MINIFLOW LINE MOV HV8813 SPURIOUSLY CLOSES DUE TO FIRE (0.28)
1-ACP-INV-MA-AD1I11__	8.81E-04	1.12E-02	1.37E+01	1.01E+00	7.96E-04	INVERTER 1AD1I11 IN MAINTENANCE
1-AFW-MDP-CO-P4003___56	5.60E-01	1.12E-02	1.01E+00	1.01E+00	1.25E-06	SPURIOUS OPERATION OF MDAFWP A (PROB 0.56)
1-EPS-DGN-FS-G4002___	2.94E-03	9.06E-03	4.07E+00	1.01E+00	1.93E-04	DG1B RANDOMLY FAILS TO START
1-ACW-MOV-OC-HV_1975__28	2.80E-01	8.04E-03	1.02E+00	1.01E+00	1.79E-06	ACCW RETURN LINE FROM RCPS MOV HV-1975 TRANSFERS CLOSED DUE TO FIRE - (PROB 0.28)
1-ACW-MOV-OC-HV_1979__28	2.80E-01	8.04E-03	1.02E+00	1.01E+00	1.79E-06	ACCW SUPPLY LINE TO RCPS MOV HV-1979 TRANSFERS CLOSED DUE TO FIRE (PROB 0.28)
1-AFW-MDP-MA-P4003___	3.00E-03	7.79E-03	3.59E+00	1.01E+00	1.62E-04	MDAFWP A (P4-003) UNAVAILABLE DUE TO T&M
1-MSS-ADV-MA-VPV3030_	3.67E-02	7.23E-03	1.19E+00	1.01E+00	1.23E-05	ARV PV3030 IN MAINTENANCE
1-ACP-INV-MA-BD1I2___	8.81E-04	6.91E-03	8.84E+00	1.01E+00	4.90E-04	INVERTER 1BD1I2 IN MAINTENANCE
1-MSS-ADV-MA-VPV3000_	3.42E-02	6.73E-03	1.19E+00	1.01E+00	1.23E-05	ARV PV3000 IN MAINTENANCE
1-LPI-MDP-FS-RHRA____	1.00E-03	6.44E-03	7.43E+00	1.01E+00	4.02E-04	RHR PUMP A RANDOMLY FAILS TO START
1-AFW-MDP-FS-P4002___	1.00E-03	5.95E-03	6.94E+00	1.01E+00	3.71E-04	MDAFWP B (P4-002) RANDOMLY FAILS TO START
1-RCS-MDP-LK-BP1	1.25E-02	5.63E-03	1.44E+00	1.01E+00	2.81E-05	RCP SEAL STAGE 1 INTEGRITY (BINDING/POPPING OPEN) FAILS
1-NSCWCT-BYPASS	9.62E-02	4.85E-03	1.05E+00	1.01E+00	3.15E-06	NSCW CTS IN BYPASS MODE (FRACTION OF TIME)
1-HPI-MOV-OC-LV0112B_28	2.80E-01	4.57E-03	1.01E+00	1.01E+00	1.02E-06	CCP RWST SUCTION ISOLATION MOV LV0112B FAILS CLOSED DUE TO FIRE (PROB 0.28)
1-SWS-MOV-MA-1668ACT_	8.73E-05	4.31E-03	5.03E+01	1.00E+00	3.08E-03	NSCW TR A RETURN ISOLATION VALVE HV1668A CLOSED FOR CT MAINT.
1-ACP-INV-FC-BD1I12__	2.15E-04	4.22E-03	2.06E+01	1.00E+00	1.23E-03	INVERTER 1BD1I12 RANDOMLY FAILS
1-ACP-INV-MA-BD1I12__	2.06E-04	4.02E-03	2.05E+01	1.00E+00	1.22E-03	INVERTER 1BD1I12 IN MAINTENANCE
1-ACP-BAC-FC-BA03____	4.78E-05	3.92E-03	8.29E+01	1.00E+00	5.12E-03	4.16KV BUS 1BA03 FAILS
1-ACP-BAC-FC-AA02____	4.78E-05	3.86E-03	8.15E+01	1.00E+00	5.03E-03	4.16KV BUS 1AA02 FAILS
1-ACP-BAC-MA-MCCBBB__	2.15E-04	3.68E-03	1.81E+01	1.00E+00	1.07E-03	480 V AC MCC 1BBB IN MAINTENANCE
1-ACP-INV-FC-AD1I1___	2.15E-04	3.62E-03	1.79E+01	1.00E+00	1.05E-03	INVERTER 1AD1I1 RANDOMLY FAILS
1-ACP-BAC-MA-MCCABB__	2.15E-04	3.13E-03	1.56E+01	1.00E+00	9.09E-04	480 V AC MCC 1ABB IN MAINTENANCE
1-EPS-TNK-MA-DFOSTKA_	6.26E-04	2.91E-03	5.65E+00	1.00E+00	2.91E-04	TR A. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-001 IN MAINTENANCE

Table 18-6 Risk Important Hardware Components

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-CVC-MDP-FR-NCP4001&	1.82E-01	2.88E-03	1.01E+00	1.00E+00	9.90E-07	NORMAL CHARGING PUMP 1208P4001 FAILS TO RUN (1 YEAR)
1-ACP-BAC-FC-AB15____	4.78E-05	2.87E-03	6.09E+01	1.00E+00	3.74E-03	480 V AC SWGR 1AB15 RANDOMLY FAILS
1-ACP-BAC-FC-BB16____	4.78E-05	2.84E-03	6.04E+01	1.00E+00	3.71E-03	480 V AC SWGR 1BB16 RANDOMLY FAILS
1-ACP-CRB-OO-AA0201__	5.35E-03	2.80E-03	1.52E+00	1.00E+00	3.27E-05	CRB AA0201 FAILS TO CLOSE ON DEMAND
1-ACP-CRB-OO-ANA0401_	5.35E-03	2.80E-03	1.52E+00	1.00E+00	3.27E-05	SAT OUTPUT CRB ANA0401 TO SAT FAILS TO CLOSE
1-ACP-INV-FC-AD1I11__	2.15E-04	2.73E-03	1.37E+01	1.00E+00	7.95E-04	INVERTER 1AD1I11 RANDOMLY FAILS
1-AFW-MDP-FS-P4003___	1.00E-03	2.72E-03	3.72E+00	1.00E+00	1.70E-04	MDAFWP A RANDOMLY FAILS TO START
1-LPI-MOV-CC-HV8811B_	3.53E-04	2.47E-03	7.99E+00	1.00E+00	4.37E-04	RHRP B CONT. SUMP Suction MOV HV8811B RANDOMLY FAILS TO OPEN
1-LPI-MOV-OO-HV8812B_	3.53E-04	2.47E-03	7.99E+00	1.00E+00	4.37E-04	RHRP B RWST SUCTION MOV HV8812B RANDOMLY FAILS TO CLOSE
1-ACP-BAC-MA-MCCBBF__	2.15E-04	2.41E-03	1.22E+01	1.00E+00	6.99E-04	480 V AC MCC 1BBF IN MAINTENANCE
1-ACP-BAC-MA-BB07____	2.15E-04	2.34E-03	1.19E+01	1.00E+00	6.79E-04	480 V AC SWGR 1BB07 IN MAINTENANCE
1-ACP-DCP-FC-1B_PS1__	1.57E-04	2.30E-03	1.56E+01	1.00E+00	9.13E-04	FAILURE OF 48V SEQUENCER POWER SUPPLY PS-1
1-ACP-DCP-FC-1B_PS4__	1.57E-04	2.30E-03	1.56E+01	1.00E+00	9.13E-04	FAILURE OF 28V SEQUENCER POWER SUPPLY PS-4
1-ACW-MOV-OC-HV_1974__28	2.80E-01	2.30E-03	1.01E+00	1.00E+00	5.13E-07	ACCW RETURN LINE FROM RCPS MOV HV-1974 TRANSFERS CLOSED DUE TO FIRE (PROB 0.28)
1-ACW-MOV-OC-HV_1978__28	2.80E-01	2.30E-03	1.01E+00	1.00E+00	5.13E-07	ACCW SUPPLY LINE TO RCPS MOV HV-1978 TRANSFERS CLOSED DUE TO FIRE (PROB 0.28)
1-ACW-MOV-OC-HV_2041_28	2.80E-01	2.30E-03	1.01E+00	1.00E+00	5.13E-07	ACCW RETURN FROM RCP TB CLG ISOLA MOV HV-2041 TRANSFERS CLOSED (0.28 - FIRE)
1-CVC-MDP-MA-CCPB____	3.00E-03	2.23E-03	1.74E+00	1.00E+00	4.65E-05	CCP-B UNAVAILABLE DUE TO MAINTENANCE
1-SWS-MOV-CC-1668A___	3.53E-04	2.22E-03	7.28E+00	1.00E+00	3.93E-04	NSCW CT A SPRAY VALVE HV1668A FAILS TO OPEN ON DEMAND
1-EPS-SEQ-FO-1821U302	3.33E-03	2.20E-03	1.66E+00	1.00E+00	4.13E-05	SEQUENCER B FAILS TO OPERATE
1-SWS-CTF-MA-_A_1234_	4.08E-05	2.10E-03	5.25E+01	1.00E+00	3.22E-03	ALL FOUR NSCW TRAIN A TOWER FANS UNAVAILABLE DUE TO MAINTENANCE (PSA VALUE)
1-SWS-MOV-CC-1669A___	3.53E-04	2.08E-03	6.89E+00	1.00E+00	3.68E-04	NSCW CT B SPRAY VALVE HV1669A FAILS TO OPEN ON DEMAND
1-AFW-MOV-OO-FV5154__	3.53E-04	2.08E-03	6.88E+00	1.00E+00	3.68E-04	MDAFWP B MINI FLOW MOV FV-5154 RANDOMLY FAILS TO CLOSE
1-SWS-CTF-MA-_B_1234_	4.08E-05	2.07E-03	5.17E+01	1.00E+00	3.17E-03	ALL FOUR NSCW TRAIN B TOWER FANS UNAVAILABLE DUE TO MAINTENANCE
1-SWS-MOV-MA-1669ACT_	4.06E-05	2.02E-03	5.08E+01	1.00E+00	3.11E-03	NSCW TR B SPRAY VALVE HV1669A CLOSED for CT MAINT.
1-AFW-TDP-FS-P4001___	5.93E-03	2.00E-03	1.33E+00	1.00E+00	2.10E-05	TDAFWP (P4-001) FAILS TO START

Table 18-6 Risk Important Hardware Components

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-HPI-MOV-OC-HV8147_28	2.80E-01	1.95E-03	1.01E+00	1.00E+00	4.34E-07	NCH ALT CHG LINE MOV HV8147 TRANSFERS CLOSED DUE TO FIRE (PROB 0.28)
1-EPS-SEQ-FO-1821U301	3.33E-03	1.93E-03	1.58E+00	1.00E+00	3.62E-05	SEQUENCER A FAILS TO OPERATE
1-CVC-MDP-TE-CCPB____	2.47E-03	1.91E-03	1.77E+00	1.00E+00	4.84E-05	CCP-B UNAVAILABLE DUE TO TEST
1-HPI-MOV-OC-HV8103C__28	2.80E-01	1.87E-03	1.01E+00	1.00E+00	4.17E-07	SPURIOUS CLOSURE OF HV8103C (DUE TO FIRE - 0.28)
1-HPI-MOV-OC-HV8103D__28	2.80E-01	1.87E-03	1.01E+00	1.00E+00	4.17E-07	SPURIOUS CLOSURE OF HV8103D (DUE TO FIRE - 0.28)
1-OEP-VCF-LP-RLOOP	1.68E-04	1.78E-03	1.16E+01	1.00E+00	6.61E-04	RANDOM LOSS OF OFFSITE POWER DURING POST-TRIP MISSION TIME (24 HOURS)
1-SWS-MDP-MA-P4_00135-3	3.39E-05	1.76E-03	5.28E+01	1.00E+00	3.24E-03	ALL 3 NSCW TRAIN A PUMPS UNAVAILABLE DUE TO MAINTENANCE
1-ACP-DCP-FC-1A_PS1__	1.57E-04	1.68E-03	1.16E+01	1.00E+00	6.65E-04	FAILURE OF 48V SEQUENCER POWER SUPPLY PS-1
1-ACP-DCP-FC-1A_PS4__	1.57E-04	1.68E-03	1.16E+01	1.00E+00	6.65E-04	FAILURE OF 28V SEQUENCER POWER SUPPLY PS-4
1-ACP-INV-FC-BD1I2___	2.15E-04	1.67E-03	8.77E+00	1.00E+00	4.85E-04	INVERTER 1BD1I2 RANDOMLY FAILS
1-DCP-BAT-MA-ND1B____	2.72E-03	1.60E-03	1.59E+00	1.00E+00	3.66E-05	125V BATTERY 1ND1B IN MAINTENANCE
1-ACP-BAC-MA-MCCBBD__	2.15E-04	1.57E-03	8.32E+00	1.00E+00	4.57E-04	480 V AC MCC 1BBD IN MAINTENANCE
1-EPS-TNK-MA-DFOSTKB_	4.00E-04	1.54E-03	4.85E+00	1.00E+00	2.41E-04	TR B. DIESEL FUEL OIL STORAGE TANK 1-2403-T4-002 IN MAINTENANCE
1-MSS-ADV-CC-VPV3000_	5.56E-03	1.51E-03	1.27E+00	1.00E+00	1.70E-05	SG1 ARV PV-3000 FAILS TO OPEN - RANDOM FAILURE
1-MSS-ADV-CC-VPV3030_	5.56E-03	1.48E-03	1.27E+00	1.00E+00	1.67E-05	SG4 ARV PV-3030 FAILS TO OPEN - RANDOM FAILURE
1-ACP-BAC-MA-MCCABF__	2.15E-04	1.48E-03	7.86E+00	1.00E+00	4.29E-04	480 V AC MCC 1ABF IN MAINTENANCE
1-ACP-BAC-MA-AB05____	2.15E-04	1.46E-03	7.81E+00	1.00E+00	4.26E-04	480 V AC SWGR 1AB05 IN MAINTENANCE
1-ESF-SSD-FC-_A518BMD	3.33E-03	1.46E-03	1.44E+00	1.00E+00	2.74E-05	DRIVER CIRCUIT ON SAFEGUARDS DRIVER CARD A518 FAILS
1-RPS-SSD-FC-__4A316B	3.33E-03	1.46E-03	1.44E+00	1.00E+00	2.74E-05	4-INPUT CIRCUIT ON UNIVERSAL LOGIC CARD A316 FAILS
1-DCP-FUS-OP-BD104___	7.46E-05	1.46E-03	2.05E+01	1.00E+00	1.22E-03	SUPPLY CURRENT FUSE BETWEEN CRB 1BD104 & INVERTER FAILS
1-UET2-NOPORV-BLK	1.10E-01	1.39E-03	1.01E+00	1.00E+00	7.92E-07	UET-ATWT
1-CVC-MDP-FS-CCPB____	1.79E-03	1.37E-03	1.77E+00	1.00E+00	4.80E-05	CCP-B FAILS TO START DUE TO RANDOM FAULTS
1-ACP-BAC-MA-AYB1____	2.15E-04	1.27E-03	6.88E+00	1.00E+00	3.68E-04	120/240V PANEL 1AYB1 IN MAINTENANCE
1-DCP-FUS-OP-AD110___	7.46E-05	1.25E-03	1.77E+01	1.00E+00	1.04E-03	SUPPLY CURRENT FUSE BETWEEN CRB 1AD110 & INVERTER FAILS
1-ACP-BAC-MA-BYB1____	2.15E-04	1.20E-03	6.57E+00	1.00E+00	3.48E-04	120/240V PANEL 1BYB1 IN MAINTENANCE
1-AFW-MDP-FR-P4002___	1.98E-04	1.16E-03	6.83E+00	1.00E+00	3.64E-04	MDAFWP B (P4-002) RANDOMLY FAILS TO RUN
1-LPI-MOV-CC-HV8811A_	3.53E-04	1.16E-03	4.27E+00	1.00E+00	2.04E-04	RHRP A CONT. SUMP SUCTION MOV HV8811A RANDOMLY FAILS TO OPEN

Table 18-6 Risk Important Hardware Components

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-LPI-MOV-OO-HV8812A_	3.53E-04	1.16E-03	4.27E+00	1.00E+00	2.04E-04	RHRP A RWST SUCTION MOV HV8812A RANDOMLY FAILS TO CLOSE
1-RPS-ICC-TE-605Q5SPB	1.34E-03	1.15E-03	1.86E+00	1.00E+00	5.36E-05	TRAIN B UNAVAILABLE FOR SEMI-AUTOMATIC LOGIC TESTING
1-CVC-MDP-MA-CCPA____	3.00E-03	1.02E-03	1.34E+00	1.00E+00	2.13E-05	CCP-A UNAVAILABLE DUE TO MAINTENANCE
1-MSS-ADV-MA-VPV3020_	1.80E-02	1.02E-03	1.06E+00	1.00E+00	3.54E-06	ARV PV3020 IN MAINTENANCE
1-AFW-TDP-MA-P4001___	3.76E-03	1.00E-03	1.27E+00	1.00E+00	1.67E-05	TDAFWP (P4-001) UNAVAILABLE DUE TO T&M

18.4.5 Key Insights of L3-FPRA Results

This section provides key insights from the development and quantification of the L3-FPRA model. Insights are provided on the following topics:

- Consequential small LOCAs
- Fire-induced and conditional LOOP
- Spurious equipment actuations and valve transfers
- Analysis of fire compartments
- Main control room
- Multi-compartment fire analysis
- Other

18.4.5.1 Consequential Small LOCAs

Consequential small LOCAs are very significant contributors to fire CDF. A consequential SLOCA can result from a stuck open or spuriously opened PORV or an RCP seal LOCA. Collectively, consequential small LOCAs contribute 22.5 percent to the total fire CDF (1.39×10^{-5}/rcy). This contribution is dominated by RCP seal LOCAs. The loss of RCP seal cooling is most likely to occur due to loss of either the auxiliary component cooling water system or nuclear service cooling water system. These two systems are also needed to support long-term reactor coolant makeup capability. Therefore, if either of these systems fails, the occurrence of an RCP seal LOCA, via either seal failure or operator failure to trip the RCPs, will lead directly to core damage. In addition, the probabilities of RCP seal failure and operator failure to trip the RCPs are relatively large (0.21 and 0.33, respectively).

18.4.5.2 Fire-Induced and Conditional LOOP

The results of the L3-FPRA indicate that fire-induced and conditional LOOP events are significant contributors to fire CDF. The fire-induced and conditional LOOP contribution is derived by gathering all cut sets that are part of the LOOPPC transfer event tree and all cut sets that contain the conditional LOOP basic events. However, due to the complexity of the modeling, the exact contribution of fire-induced and conditional LOOP to CDF is difficult to ascertain. The process used to estimate the fire-induced and conditional LOOP contribution to CDF, and its limitations, are described in more detail below.

The conditional LOOP basic events are part of the Level 1 at-power model for internal events (NRC, 2022a). These conditional LOOP basic events include a transient-induced LOOP, a LOCA-induced LOOP, and a random LOOP occurring within the mission time. If any one of these conditional LOOP events occurs, onsite emergency power is required for success. Therefore, combining these cut sets with those from the LOOPPC event tree provides a good representation of the fire-induced and conditional LOOP frequency.

The result for gathering all the cut sets that transferred through the LOOPPC event tree is 1.46×10^{-5}/rcy (23.8 percent of total fire CDF). Parsing out the cut sets that contain conditional LOOP events (OEP-VCF-LP-CLOPT [transient-induced LOOP], OEP-VCF-LP-CLOPL [LOCA-induced LOOP], and OEP-VCF-LP-RLOOP [random LOOP occurrence within the mission time]) that do not transfer through the LOOPPC event tree yields a CDF contribution of 4.87×10^{-6}/rcy (8.0 percent of total fire CDF). Adding these values together yields a fire-induced and conditional LOOP CDF of 1.95×10^{-5}/rcy (31.8 percent of total fire CDF).

The process described above only results in an approximation of the fire-induced LOOP contribution to CDF. It is difficult to truly identify which fire scenarios directly result in a LOOP event, because there are many different ways that a fire can result in a loss of offsite power (e.g., through failure of different transformers or breakers).[12] As such, there are fire-induced LOOPs that may be missed because of the location of specific fires (e.g., a fire in a 4.16 kV AC cubicle bus). These fires transfer into the LOOPPC event tree but also transfer through the other eight event trees. While the sequences in these other event trees still require onsite emergency power, they may not be picked up using the method discussed above. Although some of the fire-induced cut sets are missed with this approach, the majority of fire-induced LOOP CDF contribution is captured.

In the L3PRA at-power Level 1 PRA model for internal events, alignment of the alternate source of offsite power is credited for switchyard-centered and plant-centered LOOP events, as well as transient-induced LOOP events, given failure of onsite power (diesel generators). This same crediting of the operator action to align the alternate source of offsite power is used in the L3-FPRA, since the Level 1 at-power internal events model and assumptions are the starting point. Based on the RIR importance of the two HFEs related to the alternate source of offsite power in Table 18-5, total fire CDF would increase by approximately 49 percent if credit was not taken for aligning the alternate source of offsite power.

18.4.5.3 Spurious Equipment Actuations and Valve Transfers

Spurious equipment actuations (in particular, MSOs) that can result from a fire are important contributors to fire CDF. To obtain an estimation of the spurious operation contribution to overall fire CDF, all cut sets that contain an event that spuriously operates due to a fire are gathered into a single group and then these cut sets are re-minimalized prior to quantification. To perform this operation, SAPHIRE has features that allow the analyst to group cut sets based on characteristics or component names.

Gathering the sequence cut sets that contain a spurious operation event into a single group of cut sets is complicated by the fact that they do not go through a single event tree that is readily available for grouping. A rule was used to collect every cut set that contained a basic event that

[12] The details of these failures are contained in the flag sets used by SAPHIRE to quantify the L3-FPRA model.

involved spurious operation. This collection of cut sets has a frequency of 1.58×10^{-5}/rcy and contributes 25.8 percent to the overall fire CDF.

The component whose spurious actuation contributes the most to fire-induced spurious actuation CDF is AFW system valve. This valve is the steam supply valve to the turbine-driven AFW pump, and the spurious opening of this valve will cause the pump to start and can lead to overfilling the steam generators and an induced steam line break. This valve contributes 35.9 percent to the spurious actuation CDF. The next largest contributors to spurious actuation CDF are the power operated relief valves (PORVs). Spurious opening of the PORVs will cause a consequential LOCA (either small or medium, depending on whether one or both valves spurious open), thereby requiring reactor coolant makeup. The dominant accident sequences involving spurious operation of the PORVs also involve fire-induced damage to the makeup capability, which leads to the importance of these components. Collectively, spurious actuation of the PORVs contributes 23.1 percent to spurious actuation CDF.

All of the results discussed above looked at the spurious operation of components as a group, whether the identified component failed individually as a single spuriously operated component within a cut set or whether it was part of multiple spuriously operated components within a cut set. The overall group of cut sets involving one or more spurious operations was parsed to obtain just those cut sets that contained multiple spurious basic events. The subset of cut sets involving multiple spurious operations has a combined CDF of 1.51×10^{-6}/rcy, which contributes 9.6 percent to spurious actuation CDF. An example of multiple spurious operations that lead directly to core damage is the spurious opening of a PORV (resulting in a small LOCA) and spurious closing of the RWST suction valve for the safety injection and charging pumps.

18.4.5.4 Analysis of Fire Compartments

The RP-FPRA identified 443 fire compartments (PAUs). When ranked by CDF, the top 50 fire compartments contribute 90.1 percent of the total fire CDF in the L3-FPRA. As is typical in most fire PRAs, the fire compartments that contribute the most to CDF include the main control room, the switchgear rooms, and the cable spreading rooms.

Table 18-7 lists the fire initiating event frequency and CDF from the individual buildings and structures located at the site. The control building contributes the most to total fire CDF at 72 percent. The contributions from the other buildings to total fire CDF include: the auxiliary building (7 percent), the Unit 1 turbine building (4 percent), the Unit 1 containment building (5 percent), the Unit 2 buildings grouped together (5 percent), and the remaining buildings and yard (6 percent).

Note, from the discussion in the previous two paragraphs, the Unit 1 fire PRA includes several fire locations from Unit 2. As discussed in the RP-FPRA documentation, Unit 2 fires that can impact Unit 1 include fires originating in the shared control room or fires originating in Unit 2 PAUs that contain offsite power cables that can damage one or both transformers from Unit 1.

Table 18-7 Summary of IE and CDF Frequency in Individual Buildings

Fire Area	PAU (Fire Compartment) Description	Scenario IE Frequency (/rcy)	L3-FPRA CDF (/rcy)	L3-FPRA CDF %	No. of Scenarios
1-AB-L1	Unit 1 Auxiliary Building Level 1	2.68E-03	1.94E-06	3%	24
1-AB-L2	Unit 1 Auxiliary Building Level 2	2.21E-03	2.50E-08	0%	8
1-AB-LA	Unit 1 Auxiliary Building Level A	9.10E-04	5.02E-07	1%	10
1-AB-LB	Unit 1 Auxiliary Building Levels B	1.55E-03	6.85E-07	1%	11
1-AB-LC	Unit 1 Auxiliary Building Level C	4.54E-04	3.78E-08	0%	6
1-AB-LD	Unit 1 Auxiliary Building Level D	1.36E-02	9.10E-07	1%	70
1-AD-LB	Auxiliary Building Level B Piping Shaft	3.65E-05	2.70E-10	0%	1
	Auxiliary Building	**2.14E-02**	**4.11E-06**	**7%**	
1-AFB	Unit 1 Auxiliary Feedwater Building	1.52E-03	8.97E-07	1%	25
1-ARB-L1	Alternate Radwaste Building	5.02E-04	5.37E-09	0%	1
1-CB-L1	Control Building Level 1	1.43E-02	9.09E-06	15%	267
1-CB-L2	Control Building Level 2	2.99E-03	6.97E-06	11%	138
1-CB-L3	Control Building Level 3	5.08E-03	2.04E-07	0%	57
1-CB-L4	Control Building Level 4	2.05E-03	2.15E-08	0%	26
1-CB-LA	Unit 1 Control Building Level A	7.89E-03	2.15E-05	35%	362
1-CB-LB	Unit 1 Control Building Level B	1.05E-02	5.08E-06	8%	168
1-CB-LC	Unit 1 Control Building Level C	3.79E-03	1.55E-06	3%	57
	Control Building	**4.66E-02**	**4.44E-05**	**72%**	
1-CTB	Unit 1 Containment Building Levels C, B, A	2.39E-02	2.83E-06	5%	52
1-CWS	Unit 1 Circulating Water / Turbine Plant Cooling Water Pumps Area	1.05E-03	1.12E-08	0%	1

Table 18-7 Summary of IE and CDF Frequency in Individual Buildings

Fire Area	PAU (Fire Compartment) Description	Scenario IE Frequency (/rcy)	L3-FPRA CDF (/rcy)	L3-FPRA CDF %	No. of Scenarios
1-DB-L1	Unit 1 Diesel Generator Building	1.12E-02	1.19E-06	2%	32
1-DPB-A	Unit 1 Diesel Generator Fuel Oil Tank Level A	2.85E-04	9.24E-09	0%	1
1-DPB-B	Unit 1 Diesel Generator Fuel Oil Tank Level B	2.85E-04	9.24E-09	0%	1
	Diesel Generator Fuel Oil Tank Bldg	**5.70E-04**	**1.85E-08**	**0%**	
1-EB-B	Unit 1 Equipment Building Level 1	2.20E-04	7.21E-09	0%	1
1-FB-L3	Fuel Handling Building Level 3	1.31E-03	1.40E-08	0%	2
1-FB-LC	Fuel Handling Building Level C	7.94E-04	2.11E-08	0%	3
	Fuel Handling Bldg	**2.10E-03**	**3.51E-08**	**0%**	
1-MISC	River Intake Structure Chlorine Tank Storage Area	1.57E-02	1.71E-07	0%	4
1-NSP-LA	Unit 1 NCSW Tunnels	2.62E-03	5.23E-07	1%	20
1-RPF-L1	Radwaste Processing Facility Level 1	1.47E-03	1.57E-08	0%	1
1-RTB-L1	Radwaste Processing Facility Level 1	3.59E-05	2.52E-10	0%	4
	Radwaste Processing Bldg	**1.51E-03**	**1.60E-08**	**0%**	
1-TB	Unit 1 Turbine Building Level A	3.82E-02	2.67E-06	4%	134
2-AB-L1	Unit 2 Auxiliary Building Level 1	1.73E-03	1.84E-08	0%	14
2-AB-L2	Unit 2 Auxiliary Building Level 2	1.07E-03	1.11E-08	0%	11
2-AB-L3	Unit 2 Auxiliary Building Level 3	8.26E-04	8.83E-09	0%	1
2-AB-LA	Unit 2 Auxiliary Building Level A	9.95E-04	1.20E-08	0%	11
2-AB-LB	Unit 2 Auxiliary Building Level B	2.84E-03	3.17E-08	0%	8

Table 18-7 Summary of IE and CDF Frequency in Individual Buildings

Fire Area	PAU (Fire Compartment) Description	Scenario IE Frequency (/rcy)	L3-FPRA CDF (/rcy)	L3-FPRA CDF %	No. of Scenarios
2-AB-LC	Unit 2 Auxiliary Building Level C	4.25E-04	6.05E-09	0%	6
2-AB-LD	Unit 2 Auxiliary Building Level D	9.98E-03	1.57E-07	0%	50
2-AFB	Unit 2 Auxiliary Feedwater Building Level 1	1.51E-03	1.73E-08	0%	16
2-CB-L1	Unit 2 Control Building Levels 1	1.58E-04	1.24E-08	0%	9
2-CB-L2	Unit 2 Control Building Levels 2	7.16E-04	9.89E-08	0%	67
2-CB-L3	Unit 2 Control Building Levels 3	6.76E-05	4.81E-10	0%	9
2-CB-LA	Unit 2 Control Building Levels A	8.36E-03	6.13E-07	1%	320
2-CB-LB	Unit 2 Control Building Levels B	1.06E-02	2.46E-07	0%	131
2-CB-LC	Unit 2 Control Building Levels C	2.90E-03	5.42E-07	1%	30
2-CTB	Unit 2 Containment Building Levels C, B, A	2.38E-02	2.55E-07	0%	43
2-CWS	Unit 2 Circulating Water / Turbine Plant Cooling Water Pumps Area	1.05E-03	1.12E-08	0%	1
2-DB-L1	Unit 2 Diesel Generator Building Train B	1.14E-02	1.23E-07	0%	35
2-DPB-A	Unit 2 Diesel Generator Fuel Oil Tank Level A	2.85E-04	3.04E-09	0%	1
2-DPB-B	Unit 2 Diesel Generator Fuel Oil Tank Level B	2.85E-04	3.04E-09	0%	1
2-EB-B	Unit 2 Equipment Building Level 1	2.20E-04	2.33E-09	0%	1
2-FB-LC	Unit 2 Fuel Handling Building Levels C, B	9.97E-04	1.05E-08	0%	4
2-NSP-LA	Unit 2 Refueling Water Storage Tank	2.40E-03	2.57E-08	0%	20
2-TB	Unit 2 Turbine Building Level A	3.59E-02	1.01E-06	2%	147
	Unit 2	**1.19E-01**	**3.22E-06**	**5%**	
YARD	High Voltage Switchyard	1.31E-02	1.24E-06	2%	58

18.4.5.5 Main Control Room

During the mapping process discussed in Section 9, the 204 main control room (MCR) fire sequences modeled for Unit 1 that have a non-zero CCDP [110 MCR fire sequences originating in Unit 1 and 94 MCR fire sequences originating in Unit 2] in the RP-FPRA (from fire compartments A105-JY [Unit 1] and A105-NO [Unit 2]) were mapped into 43 different L3-FPRA fire scenarios. These MCR fire scenarios contribute 14 percent (8.58×10^{-6}/rcy) towards the total L3-FPRA fire CDF. Of the 204 MCR fire sequences in the RP-FPRA, 12 fire sequences involve MCR abandonment [6 originating in Unit 1 and 6 originating in Unit 2] and were modeled as leading directly to core damage in Unit 1. These 12 MCR abandonment fire sequences were mapped into a single L3-FPRA fire scenario (FRI-A105-JY_ABN4), which contributes less than 1 percent to total fire CDF (1.40×10^{-7}/rcy).

The modeling of MCR abandonment scenarios is based on information (assumptions and conservatisms) from the RP-FPRA documentation. The L3-FPRA model used this information directly and did not perform any independent MCR abandonment analysis.

The RP-FPRA documentation discusses the control room environment (temperature and visibility) conditions necessitating abandonment. Consistent with the MCR habitability criteria provided in NUREG/CR-6850, the RP-FPRA used a temperature exceedance criterion of 95ºC (203ºF) and an optical density exceedance criterion of 3 m^{-1}. The RP-FPRA documentation documents the control room fire modeling that was performed to obtain the threshold exceedance times and provides the times used in the RP-FPRA.

The RP-FPRA also considered the possibility that a fire could result in the loss of a sufficient set of controls to necessitate MCR abandonment. However, an evaluation by the reference plant of the impacts due to a fire in each MCR control board and control panel did not identify any scenarios that would result in the operators abandoning the MCR.

A fire in each MCR panel was analyzed to determine the probability of requiring control room abandonment. This probability is calculated by multiplying the fire non-suppression probability (based on the times for exceeding the temperature and visibility thresholds provided in a report that supports the RP-FPRA documentation) by the severity factor for each heat release rate bin (from NUREG/CR-6850). The RP-FPRA documentation also noted that each of the main control board sections has detection; therefore, an additional five minutes was added to the time calculated in the report that supports the RP-FPRA documentation.

Three different configurations were analyzed, as documented in the RP-FPRA:

1. Heating, ventilation, and air-conditioning (HVAC) in purge mode – includes operator action to place Control Building HVAC in purge mode. The human error probability (HEP) for this action was set at a screening value of 0.1. Therefore, these configurations applied a 0.9 success probability.

2. HVAC in normal mode – Considers the failure of the operator to place HVAC in purge and applied a HEP of 0.1. The 0.1 HEP bounds any random failure of the HVAC system.

3. HVAC fails – Considered only for panels ACQCBA1, BCQCBB2, and NCQHVC, which contain HVAC controls and circuits.

According to the RP-FPRA documentation, with the HVAC in purge mode, the MCR abandonment criteria are not exceeded for fixed-source fires up to the 98^{th} percentile fire. In addition, with the HVAC in purge or normal mode, the MCR abandonment criteria are not exceeded for transient combustible fires up to the 98^{th} percentile fire. By using this information, the RP-FPRA calculates abandonment probabilities for control board fires ($1.60x10^{-5}$), electrical panel fires ($2.60x10^{-6}$) and HVAC electrical panel fires ($1.82x10^{-4}$). For all MCR abandonment sequences, the RP-FPRA postulates that all equipment is failed other than those components associated with alternate shutdown capability (ASC). Given the low probability of MCR abandonment, potential for spurious operation, and IN 92-18 (loss of remote shutdown capability [NRC, 1992]) concerns, further evaluation of the ASC was not performed by the reference plant and a 1.0 CCDP was applied in the RP-FPRA (and, by extension, in the L3-FPRA). Therefore, the calculated CDF for MCR abandonment sequences was obtained by multiplying the ignition frequency by the severity factor and non-suppression probability. Also, given the MCRs for Units 1 and 2 are inter-connected, there is the potential that a fire in the Unit 2 MCR results in abandonment of the Unit 1 MCR. These scenarios are addressed in the RP-FPRA and, by extension, in the L3-FPRA.

Table 18-8 provides the results of the 12 RP-FPRA MCR fire sequences (referred to as MCR abandonment scenarios in the table) and the related L3-FPRA fire scenario.

Table 18-8 Main Control Room Abandonment Scenarios

FIRE AREA (PAU)	SCENARIO	Ignition Source	IG FREQ (/rcy)	SEVERITY FACTOR	NSP[1]	RP-FPRA (/rcy)	RP-FPRA CDF (/rcy)
A105-JY	A105-JY_ABN2	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	1.24E-04	1.82E-04	1.0	2.26E-08	2.26E-08
A105-JY	A105-JY_ABN3	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	2.48E-03	2.60E-06[2]	1.0	6.45E-09	6.45E-09
A105-JY	A105-JY_ABN4	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	8.16E-04	1.60E-05[2]	1.0	1.31E-08	1.31E-08
A105-JY	A105-JY_ABN5	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	1.24E-04	7.48E-05	1.0	9.28E-09	9.28E-09
A105-JY	A105-JY_ABN6	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	2.19E-03	2.60E-06[2]	1.0	5.69E-09	5.69E-09
A105-JY	A105-JY_ABN7	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	8.16E-04	1.60E-05[2]	1.0	1.31E-08	1.31E-08
A105-NO	A105-NO_ABN2	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	1.24E-04	1.82E-04	1.0	2.26E-08	2.26E-08
A105-NO	A105-NO_ABN3	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	2.19E-03	2.60E-06[2]	1.0	5.69E-09	5.69E-09
A105-NO	A105-NO_ABN4	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	8.16E-04	1.60E-05[2]	1.0	1.31E-08	1.31E-08
A105-NO	A105-NO_ABN5	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	1.24E-04	7.48E-05	1.0	9.28E-09	9.28E-09
A105-NO	A105-NO_ABN6	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	2.48E-03	2.60E-06[2]	1.0	6.45E-09	6.45E-09
A105-NO	A105-NO_ABN7	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	8.16E-04	1.60E-05[2]	1.0	1.31E-08	1.31E-08
L3-FPRA (single fire scenario)							
A105-JY	FRI-A105-JY_ABN4	MCR Abandonment Scenario - MCR1 MCB HVAC Normal				1.40E-07	1.40E-07

Table 18-8 Main Control Room Abandonment Scenarios

FIRE AREA (PAU)	SCENARIO	Ignition Source	IG FREQ (/rcy)	SEVERITY FACTOR	NSP[1]	RP-FPRA (/rcy)	RP-FPRA CDF (/rcy)
/rcy – per reactor critical year IG FREQ – fire sequence ignition frequency SEVERITY FACTOR – conditional probability that given a fire has occurred, it will result in target damage NSP – non-suppression probability (automatic and/or manual) RP-FPRA (/ry) – the RP-FPRA sequence frequency, which is the initiating event frequency (IGF * Severity Factor * non-suppression probability) RP-FPRA CDF – the overall sequence frequency in the RP-FPRA based on the initiating event frequency, availability factor, fire damage vector, and plant response (PRA) model (since the conditional core damage probability for all these sequences was assumed to be 1.0, the CDF is the same as the sequence frequency) Note 1. NSP is reported as 1.0 for all scenarios because the probability of non-suppression is already accounted for in the calculation of the severity factor. Note 2. The severity factor for HVAC in normal mode has been reduced by the HEP of 0.1 to account for operator failure to place the HVAC in purge mode, as discussed in the RP-FPRA documentation							

18.4.5.6 Multi-Compartment Fire Analysis Results

Section 15.4.3 described the RP-FPRA screening process for the multi-compartment fire analysis (MCA). As stated in that section, only two MCA scenarios were modeled in the L3-FPRA because the other eight fell below truncation when analyzed in the RP-FPRA, and only those RP-FPRA sequences that were above truncation were modeled in the L3-FPRA. These two scenarios are listed in Table 18-9. The CDF from these MCA scenarios is 1.26×10^{-7}/rcy, which is a small fraction of the total L3-FPRA CDF. This result supports the statement in the RP-FPRA documentation that "*the [reference plant units] are very well compartmentalized with most boundaries containing fire rated barriers. Therefore, multi compartment fires have a negligible impact on total plant risk.*"

Table 18-9 MCA Fire Scenarios Modeled in L3-FPRA

Sequence	Description	Fire Area (PAU)	L3-FPRA Scenario Name	Initiating Event (/rcy)	L3-FPRA CCDP	L3-FPRA CDF (/rcy)
TB1_A	Multi Compartment Scenario	TB1	IE-FRI-TB1_A	1.92E-06	6.48E-02	1.24E-07
TB2_A	Multi Compartment Scenario	TB2	IE-FRI-1056B-IH_TR01RR	1.92E-06	9.24E-04	1.77E-09
			Sum =	**3.84E-06**		**1.26E-07**
Sum of all 10 MCA sequences from Table 15-4			Sum=	9.68E-06		1.26E-07
			Total from all fire scenarios=			6.17E-05
/rcy = per reactor critical year						

18.4.5.7 Other

Fire scenarios that do not impact any potential mitigating equipment are assumed to cause an uncomplicated reactor trip. As such, they can be modeled using the reactor trip event tree from the Level 1 internal event PRA model. Since no mitigating equipment is impacted for these fire scenarios, it is also assumed that they will have very low CCDPs. Therefore, in the L3-FPRA, those RP-FPRA fire sequences that have very low CCDPs (i.e., less than 10^{-6}/rcy) were grouped together. A total of 481 fire sequences from the RP-FPRA meet this criterion, with a combined CDF of 5.32×10^{-9}/rcy. After being mapped to the L3-FPRA, the total CDF of these fire sequences is 4.93×10^{-8}/rcy (less than 0.1 percent of L3-FPRA fire CDF).[13]

18.4.6 Limitations and Differences

As discussed in Section 9.4.1, a mapping (grouping) approach was used to group similar RP-FPRA fire sequences together in order reduce the number of fire sequences to a manageable set of fire scenarios to be placed in the L3-FPRA model. The dominant RP-FPRA fire sequences (i.e., those with high CDF and/or high CCDP) are mapped into L3-FPRA fire

[13] The difference in CDF for these sequences between the RP-FPRA and the L3-FPRA is due to differences in the modeling of plant response in the respective reference plant and L3PRA Level 1 internal events PRA models.

scenarios either on a one-to-one basis or multiple RP-FPRA fire sequences with the same set of impacted equipment are mapped (grouped) into a single L3-FPRA fire scenario.

However, as also discussed in Section 9.4.1, the remaining 1,915 RP-FPRA fire sequences were mapped into 48 "residual" L3-FPRA fire scenarios (those labeled with "-RR"). When RP-FPRA fire sequences are grouped into a single L3-FPRA residual fire scenario, the set of impacted equipment used to evaluate that fire scenario is obtained from the RP-FPRA fire sequence with the highest CCDP that is mapped to that L3-FPRA fire scenario. While this process works well for the dominant RP-FPRA fire sequences mapped to a given L3-FPRA fire scenario, it can overestimate some of the lower RP-FPRA fire sequences. For example, if two fire sequences are grouped together and one fire sequence fails two trains of a system and the other fire sequence only fails one train of the same system, the L3-FPRA fire scenario is modeled assuming both trains are impacted by the fire.

Applying the most conservative set of component failures to all of the sequences in a residual fire scenario can lead to an over-estimation of the total CDF for that scenario; however, it is not expected that this over-estimation will significantly skew the final results, given that these sequences are generally lower contributors to fire CDF. The total CDF of the 48 residual fire scenarios is 8.43×10^{-6}/rcy, representing 13.7 percent of the total fire CDF. This same group of RP-FPRA fire sequences has a CDF of 2.88×10^{-6}/rcy, representing 7.6 percent of the total RP-FPRA fire CDF. The percent contribution from these fire scenarios in the L3-FPRA is about twice that in the RP-FPRA. This is reasonable, recognizing that this difference is due to both (1) using the highest CCDP fire sequence as the representative impact on the components modeled for that residual fire scenario and (2) differences in the corresponding internal event PRA models that form the foundation for the RP-FPRA and L3-FPRA models.

One major difference between the L3-FPRA and RP-FPRA models is associated with the data used for both human error probabilities (HEPs) and random failure rates. For example, the HEP for initiating feed and bleed is a factor of 1.7 higher in the L3-FPRA versus the RP-FPRA. This operator action is a dominant contributor for the two A105-JY fire scenarios. Another data driver is the conditional LOOP probabilities used in the two models. The RP-FPRA models a single random conditional LOOP with a probability of 8.0×10^{-5}, based on a random LOOP within the 24-hour mission time after an initiating event. This same basic event is used in the L3-FPRA and has a probability of 1.7×10^{-4}. However, the L3-FPRA also considers the possibility of a consequential LOOP following a transient or LOCA (5.3×10^{-3} and 3.0×10^{-2}, respectively[14]). This is a major contributor to the difference in CDF associated with fire scenario 1098-JD_B1.

Another major difference is that the RP-FPRA does not model the need to trip the RCPs given a loss of all seal cooling (in order to prevent an RCP seal LOCA). In the L3-FPRA, the operator action to trip the RCPs is modeled and has a failure probability of 0.33. Failure of this operator action is a dominant contributor to fire scenarios 1146-VF_TR01_RR and A105-JY_P2, contributing 48 percent and 69 percent, respectively, to the CDF for these fire scenarios.[15]

[14] The provenance of these values can be found in Section 8.2.2 of the L3PRA Level 1, at-power, internal event PRA report (NRC, 2022a).

[15] It was subsequently recognized that the 0.33 probability of the operators failing to trip the RCPs should not be applied to fire scenarios that involve loss of NSCW or ACCW (e.g., A105-JY_P2), for which a significantly lower failure probability should be used. This conservatism is anticipated to overestimate fire CDF by somewhere between 3 percent and 10 percent.

One or more of the differences identified above for the four fire scenarios with the biggest difference in CDF between the L3-FPRA and the RP-FPRA are significant contributors to many of the fire scenarios. However, these findings do not indicate major differences in the fire PRA modeling between the two studies since these differences are primarily associated with the underlying internal event PRA models.

18.4.7 Truncation Limits

The L3-FPRA results, as well as PRA results in general, are impacted by the truncation limits used during the quantification process. The truncation limit used should demonstrate convergence towards a stable result. The ASME/ANS PRA standard (ASME/ANS RA-Sa-2009, Supporting Requirement QU-B3) dictates an iterative process with convergence is considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF, and the final change is less than 5 percent. A sufficient truncation limit for a specific model is identified by the first truncation limit at which the '% Difference' column is at 5 percent or less.

SAPHIRE was modified to provide an automated process of evaluating this limit. Table 18-10 shows the results of analyzing the model at different truncation limits and the percentage difference between each truncation level. While the results in Table 18-10 indicate that a truncation limit of 10^{-11}/rcy satisfies the ASME/ANS PRA standard supporting requirement, a truncation limit of 10^{-12}/rcy was used in the L3-FPRA quantification, to be consistent with the L3PRA Level 1 internal events model quantification.

Table 18-10 Convergence Report

Truncation Limit	CDF (/rcy)	# of Cut Sets	% Difference
1.00E-08	4.29E-05	805	
1.00E-09	5.44E-05	4655	21.07%
1.00E-10	5.91E-05	20527	8.02%
1.00E-11	6.09E-05	75964	2.84%
1.00E-12	6.14E-05	232784	0.80%

18.4.8 Comparison with Other Studies

A comparison was made between the L3-FPRA CDF results and the results of several other fire PRAs. The comparison studies were NRC standardized plant analysis risk (SPAR) models, that include internal fires, for plants of similar design to the reference plant (i.e., Westinghouse PWRs). The fire CDF in these studies ranged from 5.71×10^{-5}/rcy to 1.80×10^{-4}/rcy. More detailed review of the results of these studies indicated that the dominant contributors to fire CDF were mostly LOOP/SBO-related sequences. The CDF results and dominant contributors are generally consistent with those obtained from the L3-FPRA.

19 TASK 15 – UNCERTAINTY AND SENSITIVITY ANALYSIS

19.1 Objective of the Task

The objective of Task 15 is to identify and treat uncertainty in the fire PRA, along with identifying and performing sensitivity analyses.

19.2 Reference Plant Work Performed on the Task

The reference plant's uncertainty and sensitivity analysis discusses the sensitivity of the RP-FPRA results to the key sources of uncertainty identified in NUREG/CR-6850 and discusses the results of three fire model sensitivity analyses: fire ignition frequency, not crediting equipment or systems due to unknown cable routing, and MCR abandonment. The report also provides information about the propagation of parameter uncertainty in the RP-FRPA model.

19.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the reference plant's uncertainty and sensitivity analysis. The review discusses the three sensitivity analyses performed by the reference plant and recommends that the MCR abandonment assumptions be revisited because of their impact on LERF.

19.4 L3-FPRA Approach to Address the Task

The L3PRA project team performed uncertainty and sensitivity analyses on the fire scenarios developed under Tasks 5 and 11. One motivation for performing an uncertainty analysis is to estimate the variability of the analysis results. This variability arises from uncertainties in model inputs, including basic event probabilities, initiating event frequencies, model structure, analysis assumptions, and others. In general, there are two basic types of uncertainty associated with PRAs, aleatory and epistemic. As described in NUREG-1855 (NRC, 2009a), aleatory uncertainties are associated with random variables, such as basic event probabilities and initiating event frequencies, and epistemic uncertainties are mostly associated with incompleteness in our state of knowledge for modeling plant behavior. The PRA model is an explicit model of the random processes associated with plant upset conditions and subsequent response, and thus is a model of the aleatory uncertainty. The three types of epistemic uncertainties found in a PRA are parameter uncertainty, model uncertainty, and completeness uncertainty. When an analyst indicates that they have done an uncertainty analysis in PRA, they generally mean that they evaluated the epistemic parameter uncertainty associated with the random variables included in the PRA model. The L3-FPRA logic models incorporate basic events which have uncertainties (parameter uncertainty) developed during the data derivation process. The specific parameter distributions associated with the basic events and used in the uncertainty calculations are identified in the documentation for the L3PRA Level 1 at-power model for internal events.

Model uncertainty is much more difficult to evaluate and is typically not included in PRA uncertainty calculations. In other words, the quantitative treatment of model uncertainty is not within the current PRA "state-of-practice." Completeness uncertainty, which can be thought of as a type of model uncertainty, is separately identified since it represents a type of uncertainty that cannot be quantified. Completeness uncertainty is typically addressed through a qualitative discussion of potential areas not modeled that could potentially have an impact on the final result. Since the L3-FPRA model used the information from the RP-FPRA, areas of incompleteness from that model are carried over into the L3-FPRA. The sources of uncertainty identified in the RP-FPRA do not include completeness uncertainty.

Parameter uncertainty analysis was performed for the L3-FPRA and is discussed in Section 19.4.1. Section 19.4.2 identifies and characterizes key modeling uncertainties associated with the L3-FPRA. Section 19.4.3 provides some observed insights and discusses some limited sensitivity analyses that were performed.

Beyond the limited sensitivity analyses that were performed, as discussed in Section 19.4.3, the list of modeling uncertainties in Section 19.4.2 suggests a number of other areas that are candidates for future work, though not necessarily as part of this project. A list of potential areas for future work is provided in Section 19.4.4.

19.4.1 Parameter Uncertainty Analysis

This section provides the results of the parametric uncertainty analysis performed for the internal fire CDF using the SAPHIRE software. Since the starting point of the fire PRA model was the Level 1 at-power model for internal events, the component distribution types and parameters from that model are used for the L3-FPRA parameter uncertainty analysis. For the fire scenario initiating events, a gamma distribution with 0.5 as the alpha parameter is assigned. The 0.5 alpha parameter in the gamma distribution is a broad distribution that provides a conservative representation of the uncertainty. For new components that were added to the L3-FPRA model (i.e., components that were not included in the PRA model for internal events), a constrained noninformative distribution is used. This is a distribution type that is built into SAPHIRE and is based on a beta distribution that is constrained about the mean. The beta parameter, β, is calculated internally in SAPHIRE by taking the mean and assuming the alpha parameter, α, is 0.5. The equation for determining the beta parameter given the mean and alpha parameter is:

$$\beta = \frac{\alpha}{mean}$$

The new operator actions that were added to the fire PRA model were assigned a lognormal distribution. Following the approach in (EPRI, 2013), the lognormal error factors (EFs) are based on the magnitude of the HEP, as shown below:

HEP < 0.001, EF = 10
0.001 ≤ HEP ≤ 0.3, EF = 5
0.3 < HEP ≤ 0.6, EF = 3
0.6 < HEP ≤ 0.9, EF = 2
HEP > 0.9, EF = 1

During an uncertainty analysis using Monte Carlo or Latin Hypercube sampling, it is possible to obtain samples from the tails of the distribution where the probability is negative or greater than one. Such samples are discarded in the uncertainty analysis process of SAPHIRE. This, in effect, tightens the distribution. When lognormal distributions with high mean values are used, the upper tail of the distribution may be excessively trimmed during the sampling process due to the relatively large number of discarded samples with a value greater than 1.0. This can lead to a significant lowering of the calculated mean value of the distribution.

To remedy the possibility of large decreases in the mean value (due to discarded samples), an EF threshold was applied throughout the L3PRA project. That is, if a basic event used in any of the L3PRA project models has a mean value greater than 0.2, and a lognormal distribution was assigned, the EF assigned should not be above the “EF Threshold” given in Table 19-1. If it was, the EF was reduced to the threshold value (interpolation between values may be used at the discretion of the user). This threshold value is intended to preserve the mean value and anchor the 95th percentile to a probability of approximately 0.95. Use of the threshold values reduces the number of samples discarded by SAPHIRE during uncertainty analysis but does not prevent discarded samples altogether. Also, use of the threshold values does not affect the point estimate calculations, which use the mean values.

Table 19-1 Assignment of Error Factors to Lognormal Basic Events with High Mean Values

BE Mean Value	EF Threshold	95% Probability
0.200	15.00*	0.774
0.225	15.00	0.871
0.250	11.00	0.950
0.275	6.80	0.948
0.300	5.30	0.951
0.325	4.40	0.953
0.350	3.75	0.950
0.375	3.30	0.951
0.400	2.95	0.951
0.425	2.70	0.956
0.450	2.45	0.950
0.475	2.25	0.946
0.500	2.10	0.948
0.550	1.86	0.953
0.600	1.66	0.950
0.650	1.51	0.951
0.700	1.38	0.948
0.750	1.28	0.949
0.800	1.20	0.954
0.850	1.12	0.950
0.900	1.05	0.945
>0.90	1	--

*For a probability mean value of 0.20, the EF of 15 maximizes the probability of the 95th percentile at 0.774. Higher EFs result in lower 95th percentile probabilities.

The fire PRA model was solved and the cut sets gathered into a single end state. The component parameter uncertainty for this end state was then evaluated by performing 5,000 Monte Carlo samples with a random seed number of 14237. The Monte Carlo sampling process samples a probability for each component based on its uncertainty distribution and then calculates the overall CDF. This is performed 5,000 times to obtain the uncertainty results listed below:

> 5th percentile = 2.95×10^{-5}/rcy
> 50th percentile (median) = 5.46×10^{-5}/rcy
> 95th percentile = 1.14×10^{-4}/rcy
> Mean = 6.14×10^{-5}/rcy

The results are shown graphically in Figure 19-1. Figure 19-1 is the cumulative distribution function based on the 5,000 Monte Carlo samples.[16]

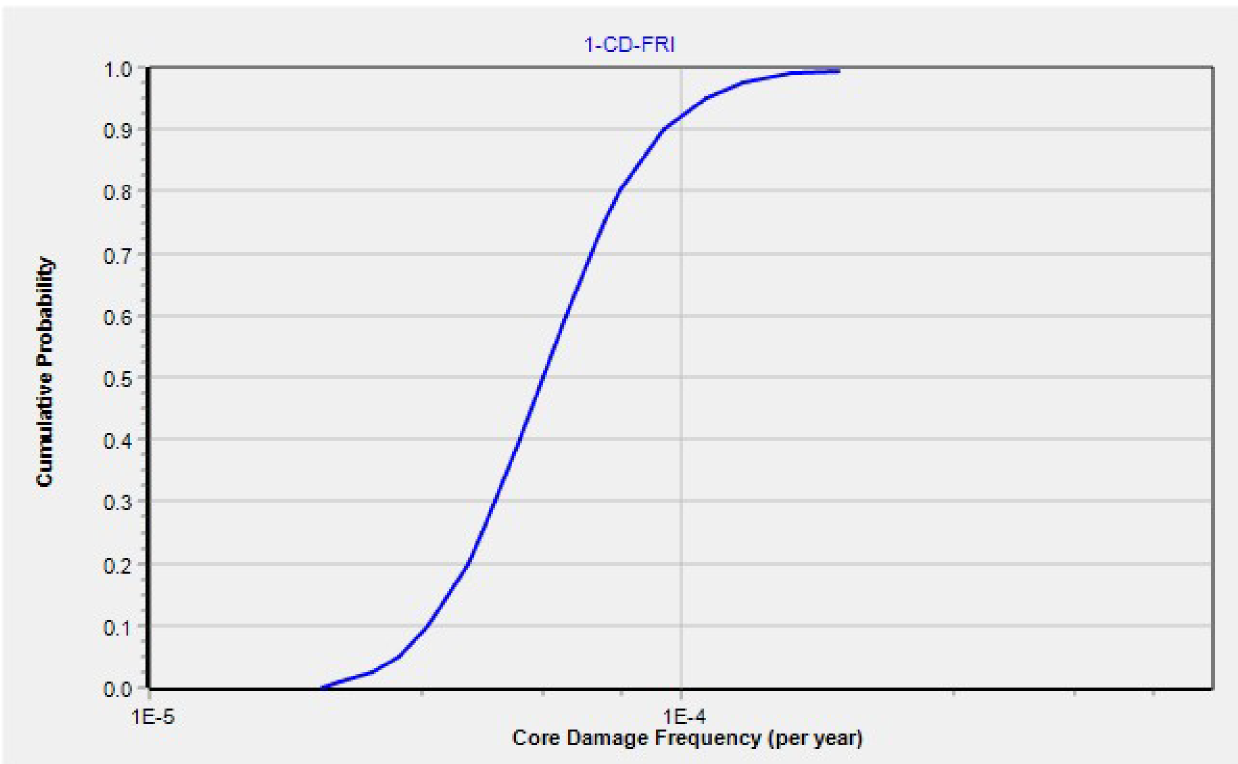

Figure 19-1 Core Damage Frequency Cumulative Distribution Function

19.4.2 Modeling Uncertainty

It is expected that modeling uncertainties, like in many other PRA models, will provide a much larger contribution to the overall CDF uncertainty than parametric uncertainties. Modeling uncertainties exist in many parts of the L3-FPRA model. The action of selecting a robust list out of many candidates itself can be viewed as posing a source of uncertainty. A list of what are considered as major modeling assumptions and uncertainties is provided in Table 19-2.

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[16] The unit of the x-axis in Figure 19-1 is per reactor critical year.

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
Initiating Event Analysis	RP-FPRA fire sequence initiating event frequencies	Combining multiple fire sequences into a single fire scenario obscures the uncertainty distributions for the individual fire sequence initiating events.	When multiple initiating events are grouped together, and assigned a group uncertainty distribution, the variability and uncertainty for the individual initiating events cannot be captured.
Accident Sequence Analysis	RP-FPRA fire sequences	The mapping process for taking the RP-FPRA sequences and combining them into single scenarios that are analyzed in the L3-FPRA.	The majority (~86 percent) of the RP-FPRA fire sequence CDF is captured in L3-FPRA scenarios either mapped on a one-to-one basis or in a group based on having the same CCDP, as discussed in Section 9.4.1. The remaining RP-FPRA sequences can be combined any of numerous ways. Based on how these sequences are grouped together can cause the final CDF to be overestimated or underestimated. This over- or under-estimation is due to the CCDP evaluated for the grouped fire scenario, which is applied to all fire sequences in the group. As stated in Section 18.4.6, when RP-FPRA fire sequences are grouped into a single L3-FPRA residual fire scenario, the set of impacted equipment used to evaluate that fire scenario is obtained from the RP-FPRA fire sequence with the highest CCDP that is mapped to that L3-FPRA fire scenario. Applying the most conservative set of component failures to all of the sequences in a residual fire scenario can lead to an over-estimation of the total CDF for that scenario; however, it is not expected that this over-estimation will significantly skew the final results, given that these sequences are generally lower contributors to fire CDF. While the grouping of residual fire scenarios as described above is not expected to have a major impact on the Level 1 fire PRA results (i.e., CDF), it is not certain

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
			if the same conclusion can be drawn for the Level 2 fire PRA. The Level 2 fire impact characterization report maps Level 2 L3-FPRA equipment failures not modeled in the Level 1 L3-FPRA to the 210 fire scenarios developed in the Level 1 L3-FPRA. For example, the L3-FPRA initiating event IE-FRI-1095-JC_F1_RR corresponds to 16 different RP-FPRA sequences, spread across 11 different fire compartments, mostly in the Unit 1 control building, but also in the auxiliary building and auxiliary feedwater building. In the fire impact characterization report, IE-FRI-1095-JC_F1_RR is marked as disabling TD-AFW blind feeding, which is not modeled in the Level 1 PRA. If some of those 16 fires would prevent blind-feed but others would not, the Level 2 L3-FPRA does not have the resolution to make that distinction. As such, the mapping process for residual scenarios may lead to some level of over- or under-estimation of release category frequencies.
	Fire scenario development	All of the fire scenarios transfer through multiple event trees to capture all potential impacts of the fire on the plant.	Evaluating all fire scenarios through multiple event trees causes a lot of duplication of cut sets; however, these cut sets on an individual sequence basis are correct. The duplication occurs because the plant response to a given fire scenario can be the same when the sequence propagates through different event trees (e.g., the transient and small LOCA event trees). While this duplication increases the complexity of the quantification process, it does not impact the total fire CDF.
	Consequential loss of offsite power (LOOP) modeling	The modeling of consequential LOOP events is included within the alternating current (AC) power fault trees for applicable	The fault tree approach used to model consequential LOOP (C-LOOP) in the Level 1 At-Power PRA model (NRC, 2022a) has

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
	(carried from internal event model)	structures, systems, and components (SSCs).	certain limitations, since C-LOOP sequences that progress to a SBO do not transfer to the SBO event tree. Therefore, aspects such as credit for AC power recovery need to be applied using post-processing rules, which can potentially miss some cut sets. Only the dominant cut sets were reviewed and had AC power recovery applied to them, which could lead to some conservatism in the final results. Note, as discussed in Section 9.4.3.2.1, the 1-FIRE-SBO node in the standard fire event tree addresses this issue for noncomplicated reactor trips (general transients). Therefore, post-processing rules are only needed for the other transfer trees (i.e., those event trees that involve some functional impact on plant response).
Success Criteria	Carried from the internal event model	See documentation for the Level 1 At-Power PRA model (NRC, 2022a).	
Systems Analysis	Spurious operation of components	Fires in certain locations can result in spurious operation of components (or multiple spurious operations).	Spurious or multiple spurious operations of components can have an impact on fire CDF. While many such spurious operations are included in the L3-FPRA model, others may not be. The L3-FPRA model relies heavily on the work already performed by the reference plant with respect to circuit analysis and spurious actuation modeling for the RP-FPRA. Spurious operations were included in the RP-FPRA based on the industry MSO expert panel approach and this work was subjected to an industry peer review. The completeness of the MSO evaluation was not verified as part of the SNL readiness review performed in support of the L3PRA project, but as stated in that review, the expert panel approach is considered acceptable by the NRC for NFPA-805

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
			application and likely captures the significant MSO cases. The probabilities of component spurious operation that are used in the RP-FPRA, which are based on the probabilities in Tables 4-1 and 4-3 of (NRC, 2014), are used directly in the L3-FPRA. According to the SNL readiness review, the reference plant used common industry modes and values and also used current reference guidance (NRC, 2014).
	Conditional system logic	System logic models are created to handle conditional effects on systems given the specific fire scenario, including the effects of fire-induced cable failures.	Fires can have multiple effects on plants and these effects need to be modeled. System logic models are created to handle the different effects (e.g., complete equipment failure or spurious operation). The fault tree logic models must be developed with sufficient detail to ensure these affects are correctly incorporated. Along with proper level of fault tree detail, cable routing needs to be understood, and the models must account for the effects of fire-induced cable failures on the systems. No cable tracing was performed specifically for the L3-FPRA, which relies on the cable tracing performed for the RP-FPRA. In instances where the RP-FPRA made assumptions in lieu of cable tracing, those assumptions were carried over to the L3-FPRA. For example, MFW and instrument air are assumed failed for all fire scenarios due to no cable tracing. One potentially important system for which no cable routing information is available is the containment spray system (which is assumed to be always failed). While this system does not influence CDF (i.e., Level 1 PRA), it can have a significant impact on the Level 2 PRA, since the

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
			operation of sprays can increase the likelihood of combustion failure of the containment. This issue is addressed more fully in the L3PRA report on the Level 2 fire PRA.
	Main Control Room Abandonment	Main control room abandonment is assumed to lead directly to core damage (CCDP = 1.0).	The L3-FPRA model uses the same MCR abandonment assumption that is used in the RP-FPRA. According to the SNL readiness review, the RP-FPRA did not credit the alternate shutdown capability given an MCR abandonment fire. The impact on the Level 1 CDF is only 2%; however, fires resulting in MCR abandonment represent approximately one-third of LERF.
	Fire spreading	Fire spreading due to heat up and other effects.	The L3-FPRA model used the same fire spread assumptions that were used in the RP-FPRA. According to the SNL readiness review, the RP-FPRA relied on judgment rather than explicit fire growth and damage modeling. The lack of rigor in the RP-FPRA approach can lead to both conservative and nonconservative outcomes. The SNL readiness review also noted that the RP-FPRA approach does not appear to include consideration of fire spread to secondary combustibles. Assumptions regarding fire spread can affect fire CDF.
Human Reliability Analysis	Operator action timing evaluations	Fires can impact the timing associated with some operator actions such that it differs from the nominal timing assumed in the internal events model.	The timing assumed for operator actions under fire conditions can have an impact on fire CDF. For the RP-FPRA, all of the operator actions in the internal event PRA model that were evaluated using the EPRI HRA Calculator were re-evaluated for the fire PRA, generally following the guidance in Appendix C of NUREG-1921 (NRC, 2012). The RP-FPRA documentation lists the various fire impacts that were considered in the re-evaluations, which include

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
			the impact of the fire on timing of (1) cues, (2) response, (3) execution, and (4) time available. These updated HEPs were used in the L3-FPRA.
	Operator ability to perform tasks	Some fires can prevent operators from performing certain actions.	Some fires are assumed to preclude certain operator actions because of the conditions the fire has created. The RP-FPRA documentation describes the initial feasibility assessment performed for operator actions in the RP-FPRA that were carried over from the internal event PRA model. Feasibility considerations included the availability of operator cues, procedure direction, personnel resources, and time for diagnosis and execution. Actions that failed the feasibility assessment were assigned an HEP of 1.0. The results of the RP-FPRA feasibility assessment were incorporated into the L3-FPRA.
Data Analysis	Fire ignition frequencies	The L3-FPRA (consistent with the RP-FPRA) uses fire ignition frequencies (FIFs) from NUREG/CR-6850, Supplement 1 (NRC, 2009c). NUREG-2169 (NRC, 2015) documents the development of updated FIFs using an enhanced methodology and incorporating updated data from EPRI's updated Fire Events Database. However, this approach could not be applied in the current study because insufficient plant-specific information was available regarding the type of ignition source for each fire in each location.	A sensitivity analysis was performed (see Section 19.4.3.8) that crudely applies the FIFs from NUREG-2169 and implies that the updated FIFs (particularly for the main control room) can significantly increase total fire CDF.
	C-LOOP probabilities (carried from internal events model)	A review of data used to estimate C-LOOP probability shows the approach has been conservative; some of the events identified as C-LOOP are actually transients with subsequent loss of offsite power resulting from random failures.	The C-LOOP probability estimate is an upper bound. The true probability is believed to be lower. Updating the C-LOOP probability estimate could have a moderate impact on total fire CDF, since C-LOOP contributes approximately 12 percent to the total.

Table 19-2 Model Assumptions and Uncertainty

Technical Element	Topic	Description	Characterization
	Sequence versus end state quantification	Quantifying the cut sets on a sequence basis versus gathering cut sets together via similar end state and then quantifying the cut sets.	The sequence quantification will evaluate each sequence independently and then sum up the overall result since the sequences are mutually exclusive. However, since the sequences from the event trees do not carry through all of the success terms (to simplify model quantification), they are not truly mutually exclusive. As such, for a given event tree, the same cut set may appear in more than one sequence, or a cut set in one sequence may be non-minimal with respect to a cut set in another sequence. This can lead to significant overestimation of CDF when summing the sequence results, particularly if the sequence cut sets involve basic events with relatively large failure probabilities (which is more common in seismic and high wind PRAs, and in Level 2 PRAs). The end state quantification gathers all of the cut sets together for each event tree, ignoring the boundaries of the individual sequences, and performs Boolean reduction on this combined set of event tree cut sets. This reduces the cut sets down to just the “minimal” ones. SAPHIRE then quantifies the event tree minimal cut sets to get event tree core damage frequency. The end state quantification approach was used to calculate the total fire CDF for the L3-FPRA.

19.4.3 L3-FPRA Insights and Sensitivity Analyses

This section provides major insights obtained during development and quantification of the L3-FPRA model and discusses the limited sensitivity analyses that were performed. The insights are associated with the following topics:

- Fire PRA realism
- Control room abandonment scenarios
- Effect of unquantified CDF cut sets with multiple human error probabilities (HEPs)
- Fire HEPs
- Fire compartment CDF comparisons
- Dominant sequence to scenario CCDP comparisons
- Effect of sequence mapping on fire CDF
- Effect of fire initiating event frequencies from 2015 NUREG-2169

19.4.3.1 Fire PRA Realism

The RP-FPRA, and by extension the L3-FPRA, were developed based on plant design and operation from circa 2012. Also, the fire PRA methods and data used for these studies represent the state-of-practice from that proximate timeframe. As such, these studies use the current guidance for performing fire PRAs, as provided in NUREG/CR-6850 (NRC, 2009c) and a series of companion documents covering various methods refinements, clarifications, and expansions developed since that publication was released, and represent an advancement in fire PRA realism since the era of NUREG-1150.

Some of the major areas of advancement in fire PRA since NUREG-1150, which are documented in NUREG/CR-6850, include:

- An in-depth review of fire events and creation of a repeatable process for classifying them. This allows for improved estimation of fire frequencies and non-suppression probabilities, based on nuclear power plant data.

- Improvements to the methods for predicting fire-induced environmental conditions, assessing the likelihood of equipment damage under these conditions, and assessing the likelihood that the fire will not be detected and suppressed before equipment damage occurs. Improvements in these areas also allow the analyst to consider the effectiveness of fire barriers in preventing fire damage to protected equipment and in preventing fire growth to neighboring compartments.

- Improvements in estimating the time-dependent temperature and heat fluxes in the neighborhood of the safety equipment of interest (i.e., the fire “targets”). This required the treatment of a variety of phenomena as the fire grows in size and severity, including the spread of fire over the initiating component, the characteristics of the fire plume and ceiling jet, the spread of the fire to non-initiating components, the development of a hot gas layer, and the propagation of the hot gas layer or fire to neighboring compartments. It also allows for an appropriate treatment of uncertainties in the structure and parameters of the models used to perform the analysis.

In addition, recent and ongoing research (performed by the NRC independently or in collaboration with EPRI) to advance understanding of fire phenomena, and to improve fire PRA methods, tools, and data, can be used to enhance fire PRA realism even beyond that reflected in the RP-FPRA and L3-FPRA. Specific examples include more realistic modeling of electrical cabinet fire growth and propagation and main control board fires, revised heat release rate distributions, and incorporation of the potential for manual fire detection and suppression prior to automatic detection.

It is unknown at present what ultimate effect these latest fire modeling improvements would have on the reported fire CDF for the L3-FPRA. However, the relatively high significance of electrical cabinet fires in the L3-FPRA results implies there could be substantial impact.

19.4.3.2 Control Room Abandonment Scenarios

The L3-FPRA modeled 12 MCR abandonment fire scenarios whose total scenario frequency is:

IE-MCR-ABANDONMENT = 1.40×10^{-7}/rcy (see Table 19-3).

The reference plant abnormal operation procedure, titled “Operation from Remote Shutdown Panels,” provides instructions on controlling and operating the plant from the shutdown panels (SDPs). These panels include SDP A, SDP B (protected), and SDP C (TDAFW). However, in the RP-PRA (and the L3-FPRA), no credit is given to use of the SDPs, and a CCDP of 1.0 is assigned to these scenarios. Thus, the total CDF from these 12 scenarios equals the total scenario frequency, that is, 1.40×10^{-7}/rcy.

For the L3-FPRA, a sensitivity analysis was performed to estimate the potential benefit of taking credit for control and operation of the plant from the remote shutdown panels,[17] per abnormal procedures, given reactor trip and MCR abandonment due to any one of the 12 scenarios.

For this purpose, the following model is defined, and its CDF is quantified:

- Fire in MCR occurs; reactor is tripped; MCR abandonment occurs. Total initiating event frequency is 1.40×10^{-7}/rcy.
- Operators control plant from the SDP using an abnormal operating procedure.
- It is assumed that only train B (protected) safety systems are available. Fail train A by setting its safety-related 4 kV AC bus basic event to TRUE.
- Use the event tree model in Figure 19-2 to model the main accident sequences.
- Introduce two new operator actions, discussed below.

Abnormal Procedure: Operation from Remote Shutdown Panels

This procedure provides operator instructions for evacuating the MCR, maintaining hot standby, and attaining cold shutdown from the SDPs. This procedure is applicable with or without the availability of offsite power. This procedure addresses the potential or actual component failures which may be induced by MCR fire events.

[17] FSAR Chapter 9: The control room complex is a shared fire area for Unit 1 and Unit 2, each having its own control area. Separate alternate safe shutdown capability is provided in the form of remote shutdown panels and other local control stations for each unit.

Table 19-3 Main Control Room Abandonment Scenarios due to Fire

Scenario	Description	Fire Area (PAU)	Ignition Frequency	Severity Factor	Non-Suppression[1]	Scenario Name	Scenario Frequency (/rcy)[2]
A105-JY_ABN4	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	A105-JY	8.16E-04	1.60E-05	1.0	IE-FRI-A105-JY_ABN4	1.31E-08
A105-JY_ABN7	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	A105-JY	8.16E-04	1.60E-05	1.0	IE-FRI-A105-JY_ABN4	1.31E-08
A105-JY_ABN3	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	A105-JY	2.48E-03	2.60E-06	1.0	IE-FRI-A105-JY_ABN4	6.45E-09
A105-JY_ABN6	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	A105-JY	2.19E-03	2.60E-06	1.0	IE-FRI-A105-JY_ABN4	5.69E-09
A105-JY_ABN2	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	A105-JY	1.24E-04	1.82E-04	1.0	IE-FRI-A105-JY_ABN4	2.26E-08
A105-JY_ABN5	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	A105-JY	1.24E-04	7.48E-05	1.0	IE-FRI-A105-JY_ABN4	9.28E-09
A105-NO_ABN4	MCR Abandonment Scenario - MCR2 MCB HVAC Normal	A105-NO	8.16E-04	1.60E-05	1.0	IE-FRI-A105-JY_ABN4	1.31E-08
A105-NO_ABN7	MCR Abandonment Scenario - MCR1 MCB HVAC Normal	A105-NO	8.16E-04	1.60E-05	1.0	IE-FRI-A105-JY_ABN4	1.31E-08
A105-NO_ABN6	MCR Abandonment Scenario - MCR1 Panels HVAC Normal	A105-NO	2.48E-03	2.60E-06	1.0	IE-FRI-A105-JY_ABN4	6.45E-09
A105-NO_ABN3	MCR Abandonment Scenario - MCR2 Panels HVAC Normal	A105-NO	2.19E-03	2.60E-06	1.0	IE-FRI-A105-JY_ABN4	5.69E-09
A105-NO_ABN2	MCR Abandonment Scenario - MCR2 Panels HVAC Fails	A105-NO	1.24E-04	1.82E-04	1.0	IE-FRI-A105-JY_ABN4	2.26E-08
A105-NO_ABN5	MCR Abandonment Scenario - MCR1 Panels HVAC Fails	A105-NO	1.24E-04	7.48E-05	1.0	IE-FRI-A105-JY_ABN4	9.28E-09
(Total) Scenario frequency (initiating event) for the sensitivity case for MCR abandonment scenarios					**Sum =**	**IE-MCR-ABANDONMENT**	**1.40E-07**

Note 1: Non-suppression is reported as 1.0 for all scenarios because the probability of non-suppression is already accounted for in the calculation of the severity factor.
Note 2: /rcy = per reactor critical year

Major Actions

- Evacuate the MCR
- Establish communications between SDPs
- Transfer controls to the SDPs
- Stabilize the plant from the SDPs
- Cooldown and depressurize the reactor coolant system (RCS) if plant conditions warrant or if the Technical Support Center decides to bring the plant to cold shutdown conditions

Staffing SDPs

A page announcement will be made that the control room is being evacuated and operators will be dispatched to the following locations:

- SDP B:
  - Shift Supervisor
  - Extra Shift Personnel
- SDP A:
  - Reactor Operator
- SDP C:
  - System Operator

Operator Actions at SDPs

All SDP operations require operator actions. Three potential operator actions at the SDPs can be envisioned for this sensitivity analysis. Two of them are credited; the third one (responding to a small LOCA [SLOCA]) is not credited, as discussed below.

1. SDP-XHE-AFW: Operators fail to provide secondary cooling at shutdown panels
2. SDP-XHE-RCSINV: Operators fail to provide RCS inventory control at shutdown panels (RCS leaks only).

Although equipment to deal with larger RCS inventory leaks (beyond chemical and volume control system [CVCS] charging capacity) exists, no credit is taken for response to a SLOCA sequence, because it does not appear to be addressed in the procedure. A SLOCA sequence may occur due to:

- RCP seal failure, leading to an RCP seal LOCA exceeding 21 gpm/pump
- Stuck open PORV or safety valve

Event Tree Success Criteria

The event tree for this process is shown in Figure 19-2. Reactor trip has already occurred; decay heat must be removed by AFW, and RCS inventory leaks must be replenished to avoid core damage. Both AFW and RCS inventory functions require operator actions:

- If secondary cooling by operator action via SDPs is successful (AFW-SDP), a SLOCA does not occur, and RCP inventory is maintained for sequences with RCS leaks within the CVCS capacity (RCS-INV), then no CD occurs. (Event Tree Sequence 1)

- If AFW-SDP fails, core damage occurs. (Event Tree Sequence 4)

- If SLOCA occurs (RCP seal LOCA >21gpm/pump or stuck open PORV or safety valve), core damage occurs. (Event Tree Sequence 3)

- If AFW-SDP is successful and a SLOCA does not occur, but RCS-INV fails (i.e., operators fail to maintain RCS inventory given RCS leaks within the CVCS capacity), core damage occurs. (Event Tree Sequence 2).

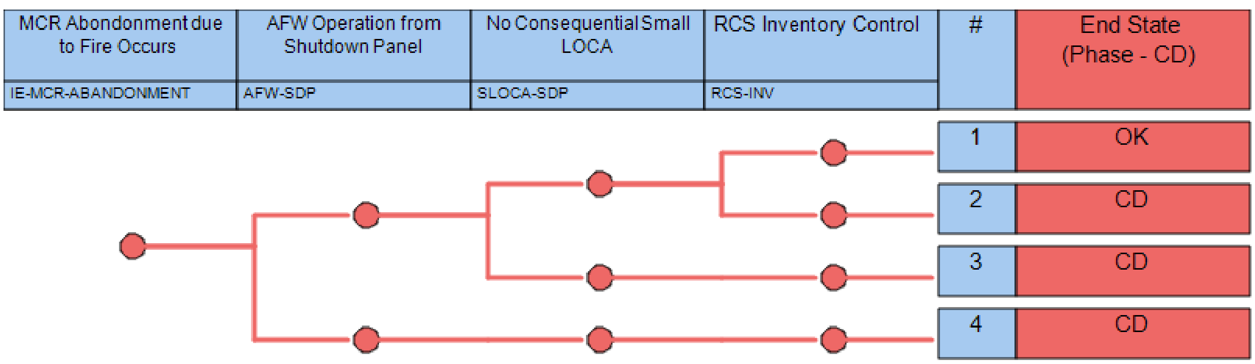

Figure 19-2 Event Tree Model for MCR Abandonment and Control from SDPs

Event Tree Nodes and Probabilities

The probabilities used below for RCP seal failure come from the WOG-2000 RCP seal failure model (NRC, 2003); the other failure probabilities are approximations from the system models or are based on analyst judgment (for the sole purpose of this sensitivity analysis).

AFW-SDP: Secondary cooling via shutdown panels (AFW). Sufficient condensate storage tank (CST) inventory exists for 24 to 72 hours.

$$P(\text{AFW-SDP}) = P(\text{SDP-XHE-AFW}) + P(\text{AFW-SDP-equipment})$$

$$P(\text{AFW-SDP}) = 0.05 + 0.002 = 0.052.$$

SLOCA-SDP: No SLOCA. Most likely source is RCP seal leaks >21 gpm/pump.

P(SLOCA-SDP) = P(RCP stage 1 or stage 2 seal failure) + P(PORV or safety valve LOCA)

P(SLOCA-SDP) = 0.21 + 0.01= 0.22 (additional spurious valve openings are not modeled).

RCS-INV: RCS inventory makeup (for RCS leaks only).

P(RCS-INV) = P(SDP-XHE-RCSINV) + P(CVCS-SDP-equipment)

P(RCS-INV) = 0.01 + 0.01 = 0.02

It is estimated that there is ample time for the operator action SDP-XHE-RCSINV; less time for SDP-XHE-AFW, if the AFW stops between reactor trip and start of SDP operations.

When the event tree is solved for core damage frequency using the MCR abandonment initiating event frequency of 1.40×10^{-7}/rcy, a CDF value of 3.85×10^{-8}/rcy is obtained, as shown in Table 19-4.

Table 19-4 Results of Main Control Room Abandonment Sensitivity Analysis

Event Tree Sequence	Sequence Frequency (/rcy)	End State	Sequence Logic
MCR-ABAND:2	2.07×10^{-9}	CD	/AFW-SDP, /SLOCA-SDP, RCS-INV
MCR-ABAND:3	2.92×10^{-8}	CD	/AFW-SDP, SLOCA-SDP
MCR-ABAND:4	7.28×10^{-9}	CD	AFW-SDP
MCR-ABAND (3 Seqs.)	3.85×10^{-8}		
/rcy = per reactor critical year			

When the CDF of $3.85 \times 10^{-8/}$rcy is compared with the base case value of 1.40×10^{-7}/rcy, a factor of 3.6 reduction in CDF is observed. Thus, taking credit for the plant control and operation from the SDP could provide a factor of 3 to 4 reduction in CDF of accident sequences with MCR abandonment due to fire.

The result is sensitive to the modeling assumption that a SLOCA occurring during these accident sequences cannot be coped with (represented by Event Tree Sequence 3). This sequence makes up roughly 75 percent of the CDF calculated with SDP credit given. If credit is given for providing make-up in response to a SLOCA, then plant control and operation from the SDP could provide a factor of more than 10 reduction in CDF of accident sequences with MCR abandonment due to fire.

19.4.3.3 Effect of Unquantified CDF Cut Sets with Multiple HEPs

In the L3-FPRA, post-processing rules are used to account for the dependencies between multiple human failure events (HFEs) occurring in the same cut set. As such, the cut sets have undergone truncation (using a truncation value of 10^{-12}/rcy) prior to the application of the post-processing rules. This can lead to some cut sets with multiple HFEs being prematurely screened out before any applicable dependencies can be accounted for. To confirm that the prematurely screened cut sets do not significantly contribute to overall fire CDF, a sensitivity analysis was performed.

To perform this sensitivity analysis, the L3-FPRA model was solved with all HEPs set to a probability of 0.9 and using a truncation level of 10^{-11}/rcy. This ensures that all cut sets containing multiple HFEs remain above truncation. The dependency rules were then applied to the resultant cut sets. These cut sets were quantified using the nominal HEPs (from Table 16-1) to obtain the final CDF result. The final CDF result with all dependencies included and using the nominal HEPs is 0.2 percent higher than that for the base case.

The small difference between the base case CDF and this sensitivity CDF does not warrant the additional analytical steps and solve time needed to make sure all dependencies are accounted for, as this will not provide any additional information. Therefore, the HFE dependency treatment in the base case is retained.

19.4.3.4 Fire HEPs

The L3-FPRA model is sensitive to HEPs in part due to the limited mitigating equipment available given a fire. From review of the top cut sets, the following four HFEs were found to be most important: (1) initiating bleed and feed operation (1-OAB_TR-------H-FIRE), (2) tripping the reactor coolant pumps (RCPs) given a loss of seal cooling (1-RCS-XHE-XM-TRIP-FIRE), (3) initiating bleed and feed operation – moderate dependence (1-OAB_TR-------H-FIRE-MD), and (4) controlling auxiliary feedwater given spurious operation (1-OACONTROL--AFW-FIRE). These four HFEs when partitioned (i.e., gathered into a single group of cut sets) contribute 34 percent to the overall fire CDF.

Importance measures were calculated for these four HFEs and are listed in Table 19-5 (a larger list is provided in Table 18-5). One of the importance measures is the risk reduction ratio (RRR). The RRR importance measure tells the analyst by what factor the overall CDF would decrease if these failure events would never occur, that is, if the operators would always perform these tasks correctly (definitions for all the importance measures are provided in Section 18.4.4). The RRR for these events is as follows: 1.16 for initiating feed and bleed (1-OAB_TR-------H-FIRE), 1.13 for tripping the reactor coolant pumps given a loss of seal cooling (1-RCS-XHE-XM-TRIP-FIRE), 1.12 for initiating feed and bleed conditioned given moderate dependence (1-OAB_TR-------H-FIRE-MD), and 1.11 for controlling auxiliary feedwater given spurious operation (1-OACONTROL--AFW).

Table 19-5 Risk Important Operator Actions

Basic Event Name	Prob	FV	RIR	RRR	Birnbaum	Description
1-OAB_TR-------H-FIRE	5.80E-02	1.36E-01	3.21E+00	1.16E+00	1.47E-04	OPERATORS FAIL TO FEED & BLEED - TRANSIENT - FIRE
1-RCS-XHE-XM-TRIP-FIRE	3.30E-01	1.12E-01	1.23E+00	1.13E+00	2.11E-05	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)
1-OAB_TR-------H-FIRE-MD	1.93E-01	1.04E-01	1.44E+00	1.12E+00	3.38E-05	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)
1-OACONTROL--AFW-FIRE	2.30E-02	9.87E-02	5.19E+00	1.11E+00	2.68E-04	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE

Sensitivity analyses were performed for the L3-FPRA by adjusting the HEP values for the different important HFEs and adjusting some of the dependencies. These sensitivity analyses were performed because of the importance of the operator actions on the overall fire PRA results. As discussed earlier, the L3-FPRA started with the HFEs from the L3PRA Level 1 PRA internal events model, and new HFEs were introduced to address conditions specific to one or more internal fire scenarios. One of the dominant HFEs in the L3-FPRA is the operator failure to trip the RCPs, which has a HEP of 0.33. This HFE is in the L3PRA Level 1 internal events model but is not modeled in the RP-FPRA (or in the reference plant internal event PRA). This operator action is a dominant contributor to many of the fire scenarios; therefore, this HFE (in combination with other HFEs) has a large impact on the overall fire CDF.[18]

Three sensitivity analyses were performed to illustrate the importance of HFEs on the L3-FPRA. The first sensitivity analysis used the RP-FPRA independent and dependent HEPs, and only included L3PRA internal event HEPs if the corresponding HFE was not included in the RP-FPRA. This analysis was used early on as an aid to understanding the modeling and for comparing the two fire PRA models (RP-FPRA and L3-FPRA). This first sensitivity also set the RCS-XHE-TRIP-FIRE to “FALSE” since it is not included in the RP-FPRA. The second sensitivity evaluated the L3-FPRA using the screening HEPs determined by SNL using NUREG-1921 (NRC, 2012), along with calculating dependencies based on these screening HEPs. Table 19-6 lists the HEPs determined for this screening. The last sensitivity analysis listed here looked at setting the RCS-XHE-XM-TRIP-FIRE to “FALSE.” This sensitivity was performed to see what the overall fire CDF would be given the same modeling assumption as in the RP-FPRA (i.e., not requiring RCP trip to prevent a seal LOCA). The results of these three sensitivity analyses are listed in Table 19-7. The table lists the overall CDF and the model of record used for each sensitivity case.

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-CAD-XHE-SAFESTBLE-FIRE	OPERATORS FAIL TO DEPRESSURIZE SECONDARY (72HR SAFE/STABLE) (FIRE RELATED)	7.50E-04	5.00E-03
1-CAD-XHE-SGTR-LT-FIRE	FAILURE TO INITIATE NORMAL COOLDOWN WITH HPI - SGTR, LATE (FIRE RELATED)	1.90E-03	5.00E-02
1-CAD-XHE-SGTR-LT-FRE-LD	FAILURE TO INITIATE NORMAL COOLDOWN WITH HPI - SGTR, LATE (FIRE RELATED - LOW DEPENDENCY)	9.75E-02	9.75E-02
1-CHG-XHE-NORMAL-FIRE	OPERATORS FAIL TO ESTABLISH CHARGING GIVEN A LOSS OF RCP SEAL INJECTION (fire related)	3.20E-04	1.00E+00
1-OA------MANRTH-FIRE	OPERATOR FAILS TO MANUALLY INITIATE A REACTOR TRIP (FIRE RELATED)	1.90E-03	5.00E-02

[18] It was subsequently recognized that the 0.33 probability of the operators failing to trip the RCPs should not be applied to fire scenarios that involve loss of NSCW or ACCW (e.g., A105-JY_P2), for which a significantly lower failure probability should be used. This conservatism is anticipated to overestimate fire CDF by somewhere between 3 percent and 10 percent.

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-OA-ALIGNPW-01HR-FIRE	OPERATORS FAIL TO ALIGN ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 1 HR AFTER SBO - FIRE	1.15E-01	1.00E+00
1-OA-ALIGNPW-02HR-FIRE	OPERATORS FAIL TO ALIGN ALTERNATE SOURCE OF OFFSITE POWER TO 4.16KV BUS WITHIN 2HR AFTER SBO - FIRE	1.22E-02	1.00E+00
1-OA-ALTAFW----H-FIRE	OPERATORS FAIL TO PROVIDE ADDITIONAL WATER SOURCE FOR LONG TERM AFW - FIRE	1.32E-03	1.00E-03
1-OA-ALTAFW----H-FIRE-LD	OPERATORS FAIL TO PROVIDE ADDITIONAL WATER SOURCE FOR LONG TERM AFW - FIRE (LOW DEPENDENCY)	5.13E-02	5.10E-02
1-OA-CCP-ALIGN---H-FIRE	OPERATORS FAIL TO SHIFT FROM NCP TO CCP AFTER LOACCW FOR RCP SL INJ. - FIRE	1.00E+00	1.00E+00
1-OA-CSISOL----H-FIRE	OPERATORS FAIS TO CLOSE CS SUCTION FROM THE RWST - FIRE	2.60E-02	5.00E-02
1-OA-CSISOL----H-FIRE-CD	OPERATORS FAIL TO CLOSE CS SUCTION FROM THE RWST - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-OA-DEP-SBO---H-FIRE	OPERATORS FAIL TO DEPRESSURIZE SG TO 300 psig IN SBO -LOCAL ARV OPERATION - FIRE	2.70E-02	1.00E+00
1-OA-ESFAS-HE1-H-FIRE	OPERATORS FAIL TO START EQUIP ON FAILURE OF ESFAS SIGNAL - FIRE	1.86E-02	5.00E-02
1-OA-ESFAS-HE1-H-FIRE-CD	OPERATORS FAIL TO START EQUIP ON FAILURE OF ESFAS SIGNAL - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-OA-HPR-ACRA--H-FIRE	OPERATORS FAIL TO SWITCH TO HPR -SBO AC recov 21/480 gpm or STKO RV w CCUs - FIRE	1.18E-03	1.00E-03
1-OA-HPRCU-ACR-H-FIRE	OPERATORS FAIL TO ESTABLISH HPR -SBO after ACR 21/182gpm w/o CCUs - FIRE	7.90E-04	1.00E-03
1-OA-HURGXFMR--H-FIRE	OPERATORS FAIL LOCAL CHANGE 120V AC SUPPLY FROM INVRTR TO RGXFMR - FIRE	8.50E-03	1.00E-01
1-OA-IS-ISLACC-H-FIRE	OPERATORS FAIL TO ISOLATE ISLOCA THROUGH ACCW RCP TB COOLING LINE - FIRE	1.00E+00	1.00E+00
1-OA-IS-ISLCP--H-FIRE	OPERATORS FAIL TO LOCATE ISLOCA PATH TO NCP/CCPS SUCTION - FIRE	1.00E+00	1.00E+00
1-OA-IS-ISLLKF-H-FIRE	OPERATORS FAIL TO ISOLATE RCP SEAL LEAK OFF ISOLATION VALVES - FIRE	1.00E+00	1.00E+00
1-OA-IS-ISLRHR-H-FIRE	OPERATORS FAIL TO ISOLATE ISLOCA THROUGH RHR CL INJ. LINES - FIRE	1.00E+00	1.00E+00
1-OA-IS-ISLSEALSBO-FIRE	OPERATORS FAIL TO ISOLATE RCP SEAL LINES at LOCAL -ISLOCA w SBO - FIRE	1.00E+00	1.00E+00
1-OA-IS-ISLSI--H-FIRE	OPERATORS FAIL TO ISOLATE ISLOCA PATH through SIS CL OR HL INJ LINES - FIRE	1.00E+00	1.00E+00
1-OA-ISL-MITI--H-FIRE	OPERATORS FAIL TO MITIGATE AND STABILIZE PLANT AFTER SMALL SIZE ISLOCA - FIRE	1.73E-04	1.00E-02
1-OA-ISOLETDOWNH-FIRE	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE	1.90E-02	5.00E-02
1-OA-ISOLETDOWNH-FIRE-CD	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-OA-LTFB-ACRA-H-FIRE	OPERATORS FAIL TO HPR FOR LONG TERM F&B -SBO after AC RECOVERY F&B inj. CCU recov - FIRE	6.00E-04	1.00E-03
1-OA-MANUAL-SI-H-FIRE	OPERATORS FAIL TO MANUALLY INITIATE A SAFETY INJECTION - FIRE	1.20E-03	1.00E-02
1-OA-N1EBATCHG-H-FIRE	OPERATORS FAIL TO PUT THE STANDBY NON 1E BATTERY CHARGER TO SERVICE - FIRE	1.00E+00	1.00E+00
1-OA-NSCWCT-MV-H-FIRE	OPERATORS FAIL TO LOCALLY OPEN NSCW CT SPRAY MOV NO SI - FIRE	1.38E-02	2.00E-02
1-OA-NSCWFAN---H-FIRE	OPERATORS FAIL TO START NSCW FAN MANUALLY (PLACE HOLDER) - FIRE	1.00E+00	1.00E+00
1-OA-OBR-------H-FIRE	OPERATORS FAIL TO ESTABLISH EMERGENCY BORATION	1.00E+00	1.00E+00
1-OA-OCR_A-----H-FIRE	OPERATORS FAIL TO STEP INSERT CONTROL RODS - FIRE	1.00E+00	1.00E+00
1-OA-OFC_1-----H-FIRE	OPERATORS FAIL TO CONTINUE TO OPERATE TDAFWP AFTER BAT DEPL -SBO w DEP failed - FIRE	3.00E-01	1.00E+00
1-OA-OFC_2-----H-FIRE	OPERATORS FAIL TO CONTINUE TDAFWP AFTER BAT DEPL. -SBO with DEP success - FIRE	3.00E-01	1.00E+00
1-OA-OLP_ML----H-FIRE	OPERATORS FAIL TO RESTART RHR PUMP FOR LPI MLOCA HPI FAILS DPI SUCC - FIRE	2.83E-02	1.00E+00
1-OA-OLP_SL----H-FIRE	OPERATORS FAIL TO RESTART RHR PUMP FOR LPI SLOCA HPI FAILS DPI SUCCESS - FIRE	1.23E-02	1.00E+00
1-OA-OLP_SL----H-FIRE-LD	OPERATORS FAIL TO RESTART RHR PUMP FOR LPI SLOCA HPI FAILS DPI SUCCESS - FIRE (LOW DEPENDENCY)	6.17E-02	1.00E+00
1-OA-OLP_STOPB-H-FIRE	OPERATORS FAIL TO STOP RHR PUMP WHEN RCS P >300 psig (WHEN CCW NOT AVAILABLE) - FIRE	1.59E-02	5.00E-02
1-OA-OP-PHASE-AH-FIRE	OPERATORS FAIL TO MANUALLY INITIATE PHASE A ISOLATION - FIRE	3.00E-03	1.00E-02
1-OA-ORS-------H-FIRE	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO (FIRE RELATED)	5.73E-02	2.50E-02
1-OA-OSW-------H-FIRE	OPERATORS FAIL TO ESTABLISH 1 NSCW PUMP FOR NSCW PUMP 1 2 3 4 5 OR 6 INITIATOR - FIRE	3.76E-02	1.00E+00
1-OA-PORVBLOCKVH-FIRE	OPERATORS FAIL TO CLOSE PRESSURIZER PORV BLOCK VALVES DURING A FIRE - FIRE	1.90E-02	5.00E-02
1-OA-SAGD-CHG--H-FIRE	OPERATORS FAIL TO ESTABLISH SAFETY GRADE CHARGING AFTER LOSINJ IE - FIRE	1.00E+00	1.00E+00
1-OA-START-ACCWH-FIRE	OPERATORS FAIL TO START ACCW PUMP FOR SPECIAL INITIATOR - FIRE	6.40E-02	5.00E-02
1-OA-START-AFW-H-FIRE	OPERATORS FAIL TO MANUALLY START AFW PUMPS IN MCR - FIRE	1.24E-02	5.00E-02
1-OA-SUMPMOV---H-FIRE	OPERATORS FAIL TO OPEN SUMP MOVS FOR RECIRCULATION -AUTO SIGNAL FAILED - FIRE	1.80E-03	5.00E-03
1-OA-SUMPMOV---H-FIRE-LD	OPERATORS FAIL TO OPEN SUMP MOVS FOR RECIRCULATION -AUTO SIGNAL FAILED DEP=LD ON OAN_SL-- - FIRE	5.08E-02	5.48E-02

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-OA-XFER-NON1EH-FIRE	OPERATORS FAIL TO ALIGN NON-1E BUSES GIVEN FAST XFER FAILS - FIRE	1.00E+00	1.00E+00
1-OA-XFER-NON1EH-LT-FIRE	OPERATORS FAIL TO ALIGN NON-1E BUSES GIVEN FAST TRANSFER FAILS - LONG-TERM - FIRE	2.70E-03	1.00E+00
1-OAB-SBOACR---H-FIRE	OPERATORS FAIL TO INITIATE FEED AND BLEED - SBO ACR	1.00E+00	1.00E+00
1-OAB_SI-------H-FIRE	OPERATORS FAIL TO BLEED & FEED -SI - FIRE	2.35E-02	1.00E+00
1-OAB_TR-------H-FIRE	OPERATORS FAIL TO FEED & BLEED - TRANSIENT - FIRE	5.80E-02	1.00E+00
1-OAB_TR-------H-FIRE-CD	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (Complete DEPENDENCY)	1.00E+00	1.00E+00
1-OAB_TR-------H-FIRE-HD	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (HIGH DEPENDENCY)	5.29E-01	1.00E+00
1-OAB_TR-------H-FIRE-MD	OPERATORS FAIL TO FEED & BLEED - TRANSIENT ON OA AFW or MFW START - FIRE (MODERATE DEPENDENCY)	1.93E-01	1.93E-01
1-OAC_AC-------H-FIRE	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED	4.38E-03	5.00E-02
1-OAC_AC-------H-FIRE-HD	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED (HIGH DEPENDENCY)	5.02E-01	5.25E-01
1-OAC_AC-------H-FIRE-LD	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED (LOW DEPENDENCY)	5.42E-02	9.75E-02
1-OAC_AC-------H-FIRE-MD	OPERATORS FAIL TO DEPRESSURIZE FOR LPI -SLOCA HPI FAILED (MODERATE DEPENDENCY)	1.47E-01	1.86E-01
1-OAC_NC-------H-FIRE	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE	2.19E-03	5.00E-02
1-OAC_NC-------H-FIRE-HD	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE (HIGH DEPENDENCY)	5.01E-01	5.25E-01
1-OAC_NC-------H-FIRE-MD	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE (MODERATE DEPENDENCY)	1.45E-01	1.86E-01
1-OAD_MLA------H-FIRE	OPERATORS FAIL TO DEPRESSURIZE SECONDARY FOR LPI - MLO w HPI FAILED - FIRE	4.44E-01	1.00E+00
1-OAD_SGR------H-FIRE	OPERATORS FAIL TO DEPRESSURIZE SECONDARY - FIRE	1.45E-03	5.00E-02
1-OAF_MFW------H-FIRE	OPERATORS FAIL TO ESTABLISH MFW TO SGs - FIRE	1.00E+00	1.00E+00
1-OAISOLSTMTDAFW-FIRE	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE	2.70E-02	1.00E-02
1-OAISOLSTMTDAFW-FIRE-CD	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-OAI_SG-------H-FIRE	OPERATORS FAIL TO ISOLATE RUPTURED SG - FIRE	2.10E-02	1.00E-02
1-OAL_LPLL-----H-FIRE	OPERATORS FAIL TO ESTABLISH LOW PRESSURE HOT LEG RECIRC - LLO - FIRE	1.30E-04	1.00E-03

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-OAN_SL-------H-FIRE	OPERATORS FAIL TO ESTABLISH NORMAL RHR -SLOCA - FIRE	1.10E-03	1.00E-03
1-OAN_SL-------H-FIRE-LD	OPERATORS FAIL TO ESTABLISH NORMAL RHR -SLOCA - FIRE (LOW DEPENDENCY)	5.10E-02	5.10E-02
1-OAR_HPATA----H-FIRE	OPERATORS FAIL TO ESTABLISH HPR DURING ATWT -WITH CCU SUCC (CS NOT ACTUATED) - FIRE	2.31E-03	1.00E-03
1-OAR_HPATB----H-FIRE	OPERATORS FAIL TO ESTABLISH HPR DURING ATWT -WITH CCU FAILED (CS ACTUATED) - FIRE	2.31E-03	1.00E-03
1-OAR_HPML-----H-FIRE	OPERATORS FAIL TO ESTABLISH HIGH PRESSURE RECIRCULATION - MLOCA (FIRE RELATED)	2.31E-03	1.00E-03
1-OAR_HPMSO----H-FIRE	OPERATORS FAIL TO ESTABLISH HPR -RWST MSO - FIRE	1.20E-02	5.00E-03
1-OAR_HPMSO----H-FIRE-LD	OPERATORS FAIL TO ESTABLISH HPR -RWST MSO - FIRE (LOW DEPENDENCY)	6.14E-02	5.48E-02
1-OAR_HPSLA----H-FIRE	OPERATORS FAIL TO ESTABLISH HPR -SLOCA with CCUs AVAILABLE - FIRE	6.00E-04	1.00E-03
1-OAR_HPSLA----H-FIRE-LD	OPERATORS FAIL TO ESTABLISH HPR -SLOCA with CCUs AVAILABLE - FIRE (LOW DEPENDENCY)	5.06E-02	5.10E-02
1-OAR_HPSLB----H-FIRE	OPERATORS FAIL TO ESTABLISH HPR -SLOCA WITH CCUs NOT AVAILABLE - FIRE	2.31E-03	1.00E-03
1-OAR_LPLL-----H-FIRE	OPERATORS FAIL TO ESTABLISH LOW PRESSURE RECIRC -LLO - FIRE	1.40E-02	5.00E-02
1-OAR_LPML-----H-FIRE	OPERATORS FAIL TO ESTABLISH LPR - MLOCA, HPI FAILED, DEP & LPI SUCCESS (FIRE RELATED)	1.50E-03	1.00E-03
1-OAR_LPSL-----H-FIRE	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION -SLOCA	1.10E-03	1.00E-02
1-OAR_LPSL-----H-FIRE-CD	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION -SLOCA (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-OAR_LPSL-----H-FIRE-LD	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION -SLOCA (LOW DEPENDENCY)	5.10E-02	5.95E-02
1-OAR_LPSL-----H-FIRE-MD	OPERATORS FAIL TO LPR AFTER DEPRESSURIZATION - SLOCA (MODERATE DEPENDENCY)	1.44E-01	1.51E-01
1-OAR_LPSL2----H-FIRE	OPERATORS FAIL TO ESTABLISH LPR AFTER DEPRESSURIZATION - SLOCA	6.80E-04	1.00E-03
1-OAR_LPSLNOHI-H-FIRE	OPERATORS FAIL TO ESTABLISH LPR -SLOCA HPI FAILED DEP for LPI & LPI SUCCESS - FIRE	3.70E-05	1.00E-03
1-OAR_LTFB-TRA-H-FIRE	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -TRANSIENT CCU AVAILABLE – FIRE	6.00E-04	1.00E-03
1-OAR_LTFB-TRA-H-FIRE-LD	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -TRANSIENT CCU AVAILABLE - FIRE (LOW DEPENDENCY)	5.06E-02	5.10E-02
1-OAR_LTFB-TRB-H-FIRE	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -TRANSIENT WITH CCU FAIL (fire related)	2.31E-03	1.00E-03

Table 19-6 Screening Human Error Probabilities

Event	Description	Baseline HEPs	Screening HEPs
1-OAR_LTFB_SLA-H-FIRE	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -SLO with CCUs - FIRE	5.80E-04	1.00E-03
1-OAT----------H-FIRE	OPERATORS FAIL TO TERMINATE SI - FIRE	6.00E-04	1.00E-02
1-OAT----------H-FIRE-CD	OPERATORS FAIL TO TERMINATE SI - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-OAT-ISINJ----H-FIRE	OPERATORS FAIL TO TERMINATE SI AFTER ISINJ INITIATING EVENT - FIRE	3.26E-04	1.00E-02
1-OAT-ISINJ----H-FIRE-CD	OPERATORS FAIL TO TERMINATE SI AFTER ISINJ INITIATING EVENT - FIRE (COMPLETE DEPENDENCY)	1.00E+00	1.00E+00
1-RCS-XHE-XM-TRIP-FIRE	OPERATORS FAIL TO TRIP REACTOR COOLANT PUMPS (FIRE RELATED)	3.30E-01	1.00E+00
1-RFL-XHE-REFILL-LT-FIRE	OPERATOR FAILS TO REFILL RWST LONG-TERM (FIRE RELATED)	1.00E-04	1.00E-03
1-RPS-XHE-XE-NSGNL-FIRE	OPERATORS FAIL TO RESPOND WITH NO RPS SIGNAL PRESENT (FIRE RELATED)	2.30E-01	1.00E+00
1-OACONTROL--AFW-FIRE	OPERATORS FAIL TO CONTROL AFW FLOW GIVEN SPURIOUS - FIRE	7.40E-02	7.40E-02
1-OA-ISOLETDOWNH-FIRE	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE	7.60E-02	7.60E-02
1-OA-PORVBLOCKVH-FIRE	OPERATORS FAIL TO CLOSE PRESSURIZER PORV BLOCK VALVES DURING A FIRE - FIRE	2.10E-02	2.10E-02
1-OAB_TR-------H-LT-FIRE	OPERATORS FAIL TO FEED & BLEED - TRANSIENT - LATE - FIRE	N/A	2.90E-03
1-OAR_LTFB_SLB-H-FIRE	OPERATORS FAIL TO ESTABLISH HPR FOR LONG TERM F&B -SLOCA WITHOUT CCUs - FIRE	N/A	2.11E-03
1-OA-CSISOL----H-FIRE-HD	OPERATORS FAIL TO CLOSE CS SUCTION FROM THE RWST - FIRE (HIGH DEPENDENCY)	N/A	5.13E-01
1-OA-ISOLETDOWNH-FIRE-HD	OPERATORS FAIL TO ISOLATE LETDOWN UPSTREAM OF RV - FIRE (HIGH DEPENDENCY)	N/A	5.10E-01
1-OAC_NC-------H-FIRE-LD	OPERATORS FAIL TO INITIATE NORMAL COOLDOWN AFTER LOCA with HPI - FIRE (LOW DEPENDENCY)	N/A	5.21E-02
1-OAISOLSTMTDAFW-FIRE-HD	OPERATORS FAIL TO ISOLATE STEAM TO THE TD AFW PUMP - FIRE (HIGH DEPENDENCY)	N/A	5.14E-01
1-OAT----------H-FIRE-HD	OPERATORS FAIL TO TERMINATE SI - FIRE (HIGH DEPENDENCY)	N/A	5.00E-01
1-OAT----------H-FIRE-LD	OPERATORS FAIL TO TERMINATE SI - FIRE (LOW DEPENDENCY)	N/A	5.06E-02
1-OAT----------H-FIRE-MD	OPERATORS FAIL TO TERMINATE SI - FIRE (MODERATE DEPENDENCY)	N/A	1.43E-01
1-OAT-ISINJ----H-FIRE-HD	OPERATORS FAIL TO TERMINATE SI AFTER ISINJ INITIATING EVENT - FIRE (HIGH DEPENDENCY)	N/A	5.00E-01

Table 19-7 Sensitivity Results of HEP Adjustments

Sensitivity Case	Sensitivity Case CDF (/rcy)	Base Case CDF (/rcy)
Level 1 HFEs along with reference plant HEPs and dependency HEPs	4.92E-05[1]	6.12E-05
Screening HEPs based on NUREG-1921	5.54E-04[1]	6.12E-05
Set RCS-XHE-XM-TRIP-FIRE to FALSE (remove this HFE from the results)	5.46E-05	6.12E-05

[1] The CDF for this case was obtained by taking the previously solved (base case) model and updating the existing cut sets using the alternative HEPs. As such, these values may be slightly underestimated, since some previously truncated cut sets may have been retained if the model had been fully requantified using the alternative (higher) HEPs.

19.4.3.5 Fire Compartment CDF Comparisons

A comparison was made of the fire compartment CDF results between the RP-FPRA and the L3-FPRA. The CDF for each fire compartment was calculated by summing up the CDFs for its respective fire sequences. In the L3-FPRA, the 2,481 RP FPRA fire sequences (those with a CDF greater than or equal to 10^{-12}/rcy) were condensed into 210 fire scenarios. To get a frequency for each of the 2,481 fire sequences to map into the fire compartments (for the L3-FPRA), the calculated conditional core damage probabilities (CCDPs) from the 210 fire scenarios were mapped back to the 2,481 fire sequences. The initiating event frequency for each fire sequence associated with a given L3-FPRA fire scenario was multiplied by the CCDP for that fire scenario to get the corresponding fire sequence CDF for the L3-FPRA. The fire sequences were then mapped back to the different fire compartments and their CDFs were summed to provide the overall fire compartment CDFs for the L3-FPRA.

A comparison between the RP-FPRA and the L3-FPRA was made for the 443 fire compartments. The ratio of the L3-FPRA CDF to the RP-FPRA CDF for the fire compartments ranges from 0.6 to 28. The fire compartments with the largest ratio are small contributors to CDF and have low CCDPs. A summation of all fire compartments that have a ratio greater than 10 shows that they cumulatively contribute less than 0.2 percent to the overall CDF. The largest ratios (differences) in fire compartment results between the RP-FPRA and the L3 FPRA arise due to the grouping of residual sequences. As discussed in Section 9.4.1, the CCDP assigned to each residual group in the L3-FPRA is obtained from the RP-FPRA fire sequence in the group that has the highest CCDP, which can lead to overestimation of the CDF for some fire sequences. When these sequences are mapped back into fire compartments, this can then lead to overestimation of fire compartment CDF.

The 10 fire compartments with the highest CDF in the L3-FPRA were compared to the corresponding compartments from the RP FPRA. The ratio (difference) between the two results varied from a factor of 1.1 to a factor of 2.9 (the compartment CDF is always higher in the L3-FPRA as compared to the RP-FPRA). Review of the fire sequences that comprise each compartment indicates that the primary differences arise from the same causes as previously discussed: (1) differences in conditional LOOP modeling and probabilities, (2) different HEP for operator action to initiate feed and bleed (1-OAB_TR-------H-FIRE), and (3) inclusion in the

L3-FPRA of the operator action to trip the RCPs given loss of all RCP seal cooling (to prevent an RCP seal LOCA).

A plot comparing the fire compartment (PAU) CDFs of the L3-FPRA to the RP-FPRA is shown in Figure 19-3. This plot shows that the L3-FPRA in general has a higher CDF; however, for the dominant fire compartments, the ratio (difference) is close to 1.0. As the fire compartment CDFs get smaller, the plot shows the differences between the L3-FPRA and RP-FPRA grow larger. These larger differences result from combining fire sequences with very small CCDPs into residual groups and then assigning these sequences the highest sequence CCDP of the group.

For example, the two PAUs highlighted in Figure 19-3, 1163-T3 and A038-G1, have the two largest PAU ratios. The PAU of 1163-T3 is based on a single RP-FPRA fire sequence and A038-G1 contains four RP-FPRA fire scenarios. In the L3-FPRA, these fire sequences have been mapped into residual fire scenarios due to their low CCDP and CDF. The RP-FPRA PAU CDF for 1163-T3 and A038-G1 are 1.27×10^{-9}/rcy and 2.73×10^{-9}/rcy, respectively. Since these two single PAUs were grouped into residual fire scenarios in the L3-FPRA, the L3-FPRA CDF is based on their starting initiating event frequency multiplied by the group CCDP. This raises the CDF for PAUs 1163-T3 and A038-G1 to 3.55×10^{-8}/rcy (ratio of 28) and 5.12×10^{-8}/rcy (ratio of 19), respectively. However, these two PAUs contribute less than 0.2 percent to the overall fire CDF.

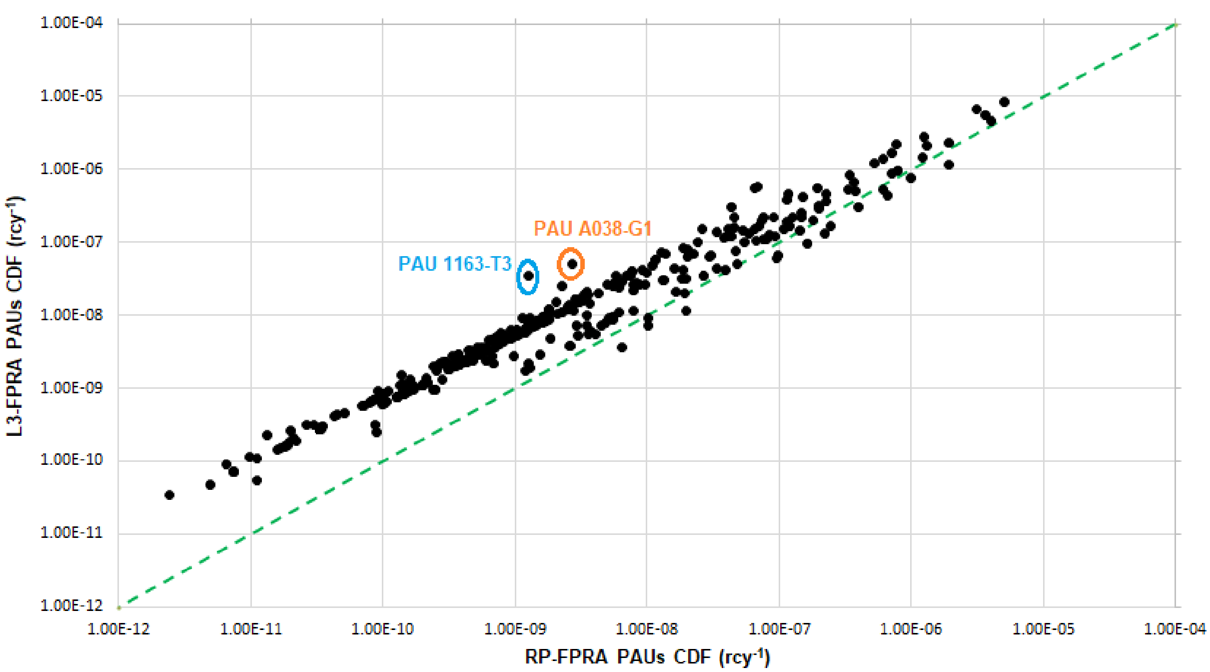

Figure 19-3 Comparison of L3-FPRA PAU to RP-FPRA PAUs

19.4.3.6 Dominant Sequence to Scenario CCDP Comparisons

When performing the comparison between the RP-FPRA and L3-FPRA models, it should be recalled that there are two distinct mapping processes: (1) direct mapping, which involves mapping of fire sequences to fire scenarios on a one-to-one basis or mapping a group of fire sequences to a single fire scenario if they occur in the same fire compartment, have the same CCDP, and impact the same safety equipment; and (2) mapping residual sequences into fire scenarios based on similar CCDP, as well as any unique nature of the fire sequences. The direct mapping created 162 unique L3-FPRA fire scenarios from 566 dominant RP-FPRA fire sequences (accounting for 86 percent of total fire CDF). The residual mapping process created 48 L3-FPRA fire scenarios from the remaining 1,915 RP-FPRA fire sequences that had a CDF greater than or equal to 10^{-12}/rcy (accounting for 14 percent of total fire CDF).

Differences between the PRA models are most easily identified when comparing those RP-FPRA fire sequences that were directly mapped to L3-FPRA fire scenarios. Reviewing the cut sets for these fire sequences reveals several modeling and data differences that drive the differences between the respective CCDPs. As discussed previously, one of those modeling differences is the inclusion in the L3-FPRA of the operator action to trip the RCPs given loss of seal cooling. The RP-FPRA does not model this operator action.

Another principal modeling difference between the two PRA models involves conditional LOOP given a transient or a LOCA. The RP-FPRA only models random conditional LOOP. In addition to random conditional LOOP, the L3-FPRA includes separate events for consequential LOOP given a transient or a LOCA, both of which have higher probabilities than that for random conditional LOOP, as discussed in Section 18.4.6.

An important data difference between the two PRA models relates to the HEP for the operator failing to initiate feed and bleed operation (1-OAB_TR-------H-FIRE). The HEP for this operator action is a factor of 1.7 higher in the L3-FPRA than in the RP-FPRA.

Lastly, due to a modeling difference in the underlying internal event PRA, the L3-FPRA does not credit recovery of common cause failure of the RAT breakers to open following a LOOP, which leads to an unrecoverable SBO.

The comparison between the RP-FPRA fire sequences and the residual L3-FPRA fire scenarios is more difficult. The residual fire scenarios are based on the target set from the RP-FPRA fire sequence that has the highest CCDP of all the fire sequences mapped to a given residual fire scenario. This allows a good comparison between the L3-FPRA fire scenario and the surrogate (i.e., representative) RP-FPRA fire sequence; however, difficulties can arise when making comparison to some of the other grouped sequences. These other fire sequences are important to the overall results for each residual fire scenario; however, they do not contribute a substantial amount to the overall fire CDF.

19.4.3.7 Effect of Sequence Mapping on Fire CDF

The mapping of thousands of fire sequences into a manageable set of fire scenarios can impact the fire CDF. For example, in the mapping process it is assumed that fire sequences that have the same CCDP fail the same target sets, which may not be completely correct. Also, the mapping of multiple fire sequences with different CCDPs into “residual” fire scenarios can introduce conservatism in the estimate of fire CDF, since the sum of the fire initiator frequency for all the grouped fire sequences is then multiplied by the CCDP of the fire scenario, which is based on the plant response for the most limiting fire sequence in the group. As such, care must be taken in mapping the fire sequences into fire scenarios, to avoid overestimating the fire risk.

The fire sequence to fire scenario mapping is relatively straight-forward for the dominant fire sequences. As mentioned previously, these were mapped on a one-to-one basis into single fire scenarios or mapped as a group of fire sequences into a single fire scenario that had the same CCDP (i.e., failed the same target set) as all of the individual fire sequences in the group. This process has no inherent issues because the target set, and hence the CCDP, is the same. Therefore, summing up the initiating event frequencies for all the RP-FPRA fire sequences mapped to a given L3-FPRA fire scenario, and multiplying by the common CCDP, will yield the same CDF as when summing the individual CDFs of all the mapped RP-FPRA fire sequences.

A minor difference in CDF can occur due to truncation when mapping fire sequences that have very close to the same CCDP. Some cut sets that would be truncated when quantifying individual fire sequences, will be retained when quantifying the associated fire scenario, since the fire scenario will have a higher initiating event frequency (due to the summation of the initiating event frequencies of the fire sequences that comprise it). Retention of these additional cut sets will result in a slightly higher CDF.

19.4.3.8 Effect of Fire Initiating Event Frequencies from NUREG-2169

NUREG-2169 (NRC, 2015) documents the development of updated fire ignition frequencies (FIFs) using an enhanced methodology and incorporating updated data from EPRI's updated Fire Events Database. This study evaluated FIFs for different fire types within different building locations (e.g., cable fires caused by welding and cutting in a Control/Auxiliary/Reactor Building). Table 4-6 of NUREG-2169 provides the updated FIFs, along the corresponding FIFs from NUREG/CR-6850, Supplement 1 (NRC, 2009c). NUREG-2169 documented that the overall FIF increased by an average of 36 percent when compared to the FIFs reported in NUREG/CR-6850, Supplement 1. This is based on some FIFs increasing and some decreasing, depending on how data from the 1990s was used. If the overall average increase in FIF is simply applied to the nominal CDF from the L3-FPRA, the total fire CDF increases from 6.14×10^{-5}/rcy to 8.35×10^{-5}/rcy.

Another approach to perform this sensitivity analysis would be to look at the individual ratios between the updated FIFs in NUREG-2169 and the corresponding FIFs in NUREG/CR-6850, Supplement 1. Table 4-6 of NUREG-2169 provides a list of the updated FIFs based on location and ignition source, along with the representative FIF from NUREG/CR-6850, Supplement 1. These ratios can be applied to the FIFs provided in the RP-FPRA based on location. However, the RP-FPRA does not provide the actual ignition source and only provides a description of the fire location. Therefore, performing this sensitivity analysis approach requires making additional assumptions.

The principal assumption to be made is what ratios should be used for the high-level locations listed in Table 4-6 of NUREG-2169. One approach is to take the average of the ratios for each location. For example, for "Control/Auxiliary/Reactor Building" three different ratios are provided based on fire ignition source. These ratios are 0.63, 1.80, and 0.69 for cable fires caused by welding and cutting, transient fires caused by welding and cutting, and transient combustibles, respectively. The average of these is 1.04; therefore, this ratio could be used for all fire locations within the control, auxiliary, and reactor buildings. This same process could be performed for the remaining locations identified. The average ratios using this approach are listed in Table 19-8.

For each L3-FPRA fire scenario, Table 19-8 provides the updated FIFs (i.e., ratio times FIF provided by RP-FPRA), the ratio of the updated FIFs to the nominal FIFs, the updated (sensitivity case) CDF, the nominal (base case) CDF, and the change in CDF, when solving the model at a truncation of 10^{-12}/rcy. The use of the updated FIFs from NUREG-2169 results in an overall increase in total fire CDF of approximately 75 percent (from 6.14×10^{-5}/rcy to 1.07×10^{-4}/rcy). Examination of the results in Table 19-8 shows that the vast majority of the CDF increase comes from fires in the control room, where the "average" FIF is increased by a factor of 5.96.

As noted above, the lack of plant-specific information regarding the type of ignition source for each fire in each location led to the use of a crude averaging approach for this sensitivity analysis. While the results of this sensitivity analysis indicate that the more recent FIFs can lead to a significant increase in estimated fire CDF, it should be recognized that the resulting values are subject to great uncertainty due to the crude averaging approach that was applied.

Table 19-8 Delta CDF of Fire Ignition Frequency Sensitivity

L3-FPRA Fire Scenario	Generic Fire Location	Ratio[2]	Updated FIF[1]	Sensitivity CDF (/rcy)	Nominal CDF (/rcy)	Delta CDF
IE-FRI-1002-AB_B0	Control/Aux/Rx Building	1.04	4.38E-04	2.14E-07	2.06E-07	8.48E-09
IE-FRI-1011A-CE_TR01	Control/Aux/Rx Building	1.04	4.43E-05	2.28E-07	2.19E-07	9.03E-09
IE-FRI-1011B-A1_TR01	Control/Aux/Rx Building	1.04	4.44E-06	1.18E-07	1.13E-07	4.65E-09
IE-FRI-1014D-B9_A_RR	Control/Aux/Rx Building	1.04	8.68E-04	2.96E-08	2.84E-08	1.17E-09
IE-FRI-1016-AV_A_RR	Control/Aux/Rx Building	1.04	1.46E-02	1.99E-07	1.91E-07	7.86E-09
IE-FRI-1017-AW_TR02_RR	Control/Aux/Rx Building	1.04	1.63E-07	1.11E-09	1.06E-09	4.38E-11
IE-FRI-1023-B6_TR01	Control/Aux/Rx Building	1.04	8.92E-05	5.53E-07	5.31E-07	2.19E-08
IE-FRI-1025-BT_A_RR	Control/Aux/Rx Building	1.04	4.64E-03	1.52E-07	1.46E-07	6.01E-09
IE-FRI-1026A-C7_A_RR	Control/Aux/Rx Building	1.04	6.61E-04	3.97E-08	3.81E-08	1.57E-09
IE-FRI-1030-C7_A_RR	Control/Aux/Rx Building	1.04	1.12E-02	1.70E-07	1.63E-07	6.71E-09
IE-FRI-1031-C6_A_RR	Control/Aux/Rx Building	1.04	2.65E-03	8.59E-08	8.25E-08	3.40E-09
IE-FRI-1039C-CU_TR01	Control/Aux/Rx Building	1.04	4.19E-05	1.62E-07	1.55E-07	6.39E-09
IE-FRI-1042B-I1_TR03	Control/Aux/Rx Building	1.04	3.35E-05	2.19E-07	2.10E-07	8.65E-09
IE-FRI-1043-D1_B0	Control/Aux/Rx Building	1.04	5.81E-04	4.70E-07	4.51E-07	1.86E-08
IE-FRI-1044-D2_B0	Control/Aux/Rx Building	1.04	4.65E-04	7.25E-07	6.96E-07	2.87E-08
IE-FRI-1044-D2_B1	Control/Aux/Rx Building	1.04	1.16E-04	5.46E-07	5.25E-07	2.16E-08
IE-FRI-1048-DC_TR01	Control/Aux/Rx Building	1.04	3.94E-05	2.03E-07	1.95E-07	8.02E-09
IE-FRI-1056A-IM_TR01RR	Control/Aux/Rx Building	1.04	4.62E-06	3.20E-09	3.08E-09	1.27E-10
IE-FRI-1056B-IH_TR01RR	Battery Room	0.60	3.67E-05	3.39E-08	5.64E-08	-2.25E-08
IE-FRI-1059-JR_A_RR	Control/Aux/Rx Building	1.04	4.43E-03	3.28E-08	3.15E-08	1.30E-09
IE-FRI-1062-JM_TR09	Control/Aux/Rx Building	1.04	1.59E-06	1.95E-08	1.87E-08	7.70E-10
IE-FRI-1066-IA_TR02	Control/Aux/Rx Building	1.04	1.94E-07	7.32E-09	7.04E-09	2.90E-10
IE-FRI-1071-IF_G1_RR	Control/Aux/Rx Building	1.04	1.19E-05	5.32E-09	5.11E-09	2.11E-10
IE-FRI-1073-I7_TR03RR	Control/Aux/Rx Building	1.04	9.89E-06	6.39E-08	6.14E-08	2.53E-09
IE-FRI-1074-ID_B1	Control/Aux/Rx Building	1.04	9.95E-05	2.51E-07	2.41E-07	9.93E-09
IE-FRI-1074-ID_E	Control/Aux/Rx Building	1.04	2.07E-04	6.18E-07	5.94E-07	2.45E-08
IE-FRI-1075-I8_C01	Control/Aux/Rx Building	1.04	1.45E-04	9.90E-07	9.51E-07	3.92E-08
IE-FRI-1075-I8_D01	Control/Aux/Rx Building	1.04	5.63E-05	3.71E-07	3.57E-07	1.47E-08

Table 19-8 Delta CDF of Fire Ignition Frequency Sensitivity

L3-FPRA Fire Scenario	Generic Fire Location	Ratio[2]	Updated FIF[1]	Sensitivity CDF (/rcy)	Nominal CDF (/rcy)	Delta CDF
IE-FRI-1075-I8_E1	Control/Aux/Rx Building	1.04	1.16E-04	6.53E-07	6.27E-07	2.58E-08
IE-FRI-1075-I8_F01	Control/Aux/Rx Building	1.04	1.45E-04	7.95E-07	7.64E-07	3.15E-08
IE-FRI-1076-IC_D	Control/Aux/Rx Building	1.04	4.14E-05	6.83E-08	6.56E-08	2.70E-09
IE-FRI-1076-IC_I	Control/Aux/Rx Building	1.04	4.14E-05	6.83E-08	6.56E-08	2.70E-09
IE-FRI-1077A-IJ_B1	Control/Aux/Rx Building	1.04	2.25E-05	9.04E-08	8.69E-08	3.58E-09
IE-FRI-1077A-IJ_TR01RR	Control/Aux/Rx Building	1.04	2.02E-06	7.84E-09	7.53E-09	3.10E-10
IE-FRI-1078A-IL_C1	Control/Aux/Rx Building	1.04	1.40E-05	1.14E-07	1.10E-07	4.53E-09
IE-FRI-1078A-IL_G_RR	Control/Aux/Rx Building	1.04	9.47E-04	1.50E-06	1.44E-06	5.94E-08
IE-FRI-1078A-IL_TR01	Control/Aux/Rx Building	1.04	1.16E-07	8.21E-10	7.89E-10	3.25E-11
IE-FRI-1079A-I9_B1	Control/Aux/Rx Building	1.04	2.56E-05	1.92E-07	1.85E-07	7.61E-09
IE-FRI-1079A-I9_TR01	Control/Aux/Rx Building	1.04	4.82E-07	3.47E-09	3.33E-09	1.37E-10
IE-FRI-1080-IS_B1	Control/Aux/Rx Building	1.04	9.95E-05	1.66E-07	1.60E-07	6.57E-09
IE-FRI-1080-IS_G2	Control/Aux/Rx Building	1.04	5.43E-05	8.95E-08	8.60E-08	3.54E-09
IE-FRI-1080-IS_H2	Control/Aux/Rx Building	1.04	1.93E-04	3.23E-07	3.10E-07	1.28E-08
IE-FRI-1080-IS_K2	Control/Aux/Rx Building	1.04	3.63E-04	6.10E-07	5.86E-07	2.41E-08
IE-FRI-1083-IG_TR01	Control/Aux/Rx Building	1.04	3.80E-05	6.30E-08	6.05E-08	2.49E-09
IE-FRI-1085-JF_TR01	Control/Aux/Rx Building	1.04	3.83E-06	1.50E-07	1.44E-07	5.92E-09
IE-FRI-1086-KB_A_RR	Control/Aux/Rx Building	1.04	4.94E-04	1.85E-08	1.78E-08	7.31E-10
IE-FRI-1091-J8_B100	Control/Aux/Rx Building	1.04	8.09E-05	1.84E-06	1.77E-06	7.29E-08
IE-FRI-1091-J8_B104	Control/Aux/Rx Building	1.04	3.64E-04	1.61E-06	1.55E-06	6.38E-08
IE-FRI-1091-J8_B200	Control/Aux/Rx Building	1.04	7.52E-05	1.25E-06	1.20E-06	4.96E-08
IE-FRI-1091-J8_B212	Control/Aux/Rx Building	1.04	5.18E-04	8.26E-07	7.94E-07	3.27E-08
IE-FRI-1091-J8_B3	Control/Aux/Rx Building	1.04	4.27E-07	1.29E-08	1.24E-08	5.11E-10
IE-FRI-1091-J8_C0	Control/Aux/Rx Building	1.04	2.07E-04	8.48E-07	8.15E-07	3.36E-08
IE-FRI-1091-J8_E0	Control/Aux/Rx Building	1.04	1.91E-05	4.33E-07	4.16E-07	1.71E-08
IE-FRI-1091-J8_E2	Control/Aux/Rx Building	1.04	1.20E-07	3.55E-09	3.41E-09	1.40E-10
IE-FRI-1092-J9_C100	Control/Aux/Rx Building	1.04	1.06E-04	5.84E-07	5.61E-07	2.31E-08
IE-FRI-1092-J9_C104	Control/Aux/Rx Building	1.04	4.69E-05	1.74E-06	1.67E-06	6.88E-08
IE-FRI-1092-J9_C113	Control/Aux/Rx Building	1.04	1.17E-04	6.30E-07	6.05E-07	2.49E-08
IE-FRI-1092-J9_C204	Control/Aux/Rx Building	1.04	1.01E-04	1.61E-06	1.55E-06	6.39E-08
IE-FRI-1092-J9_C221	Control/Aux/Rx Building	1.04	6.71E-04	4.86E-07	4.67E-07	1.92E-08
IE-FRI-1092-J9_C3	Control/Aux/Rx Building	1.04	4.43E-07	1.63E-08	1.56E-08	6.44E-10
IE-FRI-1092-J9_D0	Control/Aux/Rx Building	1.04	2.42E-05	1.33E-07	1.28E-07	5.26E-09
IE-FRI-1092-J9_D1_RR	Control/Aux/Rx Building	1.04	5.08E-04	5.70E-09	5.48E-09	2.26E-10
IE-FRI-1092-J9_E0	Control/Aux/Rx Building	1.04	1.21E-04	6.54E-07	6.28E-07	2.59E-08
IE-FRI-1093-JA_TR02	Control/Aux/Rx Building	1.04	8.69E-07	3.18E-08	3.05E-08	1.26E-09
IE-FRI-1093-JA_TR03	Control/Aux/Rx Building	1.04	8.69E-07	2.21E-08	2.12E-08	8.75E-10
IE-FRI-1093-JA_TR04	Control/Aux/Rx Building	1.04	6.08E-06	4.97E-07	4.78E-07	1.97E-08
IE-FRI-1094-KQ_B1	Control/Aux/Rx Building	1.04	8.29E-05	1.26E-06	1.21E-06	4.98E-08
IE-FRI-1094-KQ_C1	Control/Aux/Rx Building	1.04	8.29E-05	7.44E-08	7.14E-08	2.94E-09
IE-FRI-1094-KQ_H1	Control/Aux/Rx Building	1.04	5.53E-05	2.26E-07	2.17E-07	8.92E-09
IE-FRI-1094-KQ_J1	Control/Aux/Rx Building	1.04	5.53E-05	1.38E-07	1.33E-07	5.48E-09
IE-FRI-1094-KQ_TR03	Control/Aux/Rx Building	1.04	3.80E-08	2.53E-09	2.43E-09	1.00E-10
IE-FRI-1095-JC_B5	Control/Aux/Rx Building	1.04	8.04E-06	5.92E-08	5.68E-08	2.34E-09
IE-FRI-1095-JC_B8	Control/Aux/Rx Building	1.04	4.64E-07	3.54E-07	3.40E-07	1.40E-08
IE-FRI-1095-JC_D5	Control/Aux/Rx Building	1.04	4.02E-06	3.59E-08	3.45E-08	1.42E-09
IE-FRI-1095-JC_E3	Control/Aux/Rx Building	1.04	2.35E-05	2.39E-07	2.29E-07	9.45E-09

Table 19-8 Delta CDF of Fire Ignition Frequency Sensitivity

L3-FPRA Fire Scenario	Generic Fire Location	Ratio[2]	Updated FIF[1]	Sensitivity CDF (/rcy)	Nominal CDF (/rcy)	Delta CDF
IE-FRI-1095-JC_E7	Control/Aux/Rx Building	1.04	2.65E-06	1.20E-07	1.15E-07	4.74E-09
IE-FRI-1095-JC_F1_RR	Control/Aux/Rx Building	1.04	3.92E-04	9.62E-08	9.24E-08	3.81E-09
IE-FRI-1095-JC_F4	Control/Aux/Rx Building	1.04	2.49E-05	3.78E-07	3.63E-07	1.50E-08
IE-FRI-1095-JC_G1	Control/Aux/Rx Building	1.04	4.41E-05	2.33E-07	2.24E-07	9.24E-09
IE-FRI-1095-JC_G3	Control/Aux/Rx Building	1.04	1.95E-05	2.90E-07	2.78E-07	1.15E-08
IE-FRI-1095-JC_G5	Control/Aux/Rx Building	1.04	4.02E-06	6.90E-08	6.63E-08	2.73E-09
IE-FRI-1095-JC_G7	Control/Aux/Rx Building	1.04	1.33E-06	6.67E-08	6.40E-08	2.64E-09
IE-FRI-1095-JC_J3	Control/Aux/Rx Building	1.04	8.28E-05	2.82E-07	2.71E-07	1.11E-08
IE-FRI-1095-JC_K1	Control/Aux/Rx Building	1.04	6.43E-05	1.93E-07	1.85E-07	7.63E-09
IE-FRI-1095-JC_N3_RR	Control/Aux/Rx Building	1.04	1.44E-04	6.53E-08	6.28E-08	2.59E-09
IE-FRI-1097-JJ_TR01	Control/Aux/Rx Building	1.04	5.27E-06	3.41E-07	3.27E-07	1.35E-08
IE-FRI-1097-JJ_TR03	Control/Aux/Rx Building	1.04	5.27E-06	8.52E-08	8.19E-08	3.37E-09
IE-FRI-1097-JJ_TR04	Control/Aux/Rx Building	1.04	5.55E-07	3.17E-08	3.04E-08	1.25E-09
IE-FRI-1098-JD_B1	Control/Aux/Rx Building	1.04	2.49E-04	2.20E-06	2.12E-06	8.72E-08
IE-FRI-1098-JD_TR01	Control/Aux/Rx Building	1.04	3.80E-08	3.88E-10	3.73E-10	1.54E-11
IE-FRI-1099-J5_A_RR	Control/Aux/Rx Building	1.04	9.57E-04	3.18E-07	3.05E-07	1.26E-08
IE-FRI-1103-J8_B1	Control/Aux/Rx Building	1.04	2.49E-04	1.54E-06	1.48E-06	6.08E-08
IE-FRI-1103-J8_TR01	Control/Aux/Rx Building	1.04	3.07E-07	1.14E-08	1.09E-08	4.51E-10
IE-FRI-1113-JZ_TR03_RR	Plant Wide Buildings	1.92	1.22E-05	5.41E-09	2.81E-09	2.59E-09
IE-FRI-1120-KH_C3	Control/Aux/Rx Building	1.04	1.46E-05	4.19E-07	4.02E-07	1.66E-08
IE-FRI-1120-KH_C4	Control/Aux/Rx Building	1.04	1.10E-05	8.47E-07	8.13E-07	3.35E-08
IE-FRI-1120-KH_C6	Control/Aux/Rx Building	1.04	1.54E-06	7.41E-07	7.11E-07	2.93E-08
IE-FRI-1120-KH_E4	Control/Aux/Rx Building	1.04	8.45E-06	3.83E-07	3.68E-07	1.51E-08
IE-FRI-1120-KH_E5	Control/Aux/Rx Building	1.04	2.57E-06	1.96E-07	1.88E-07	7.76E-09
IE-FRI-1120-KH_F3	Control/Aux/Rx Building	1.04	7.29E-06	6.40E-08	6.15E-08	2.53E-09
IE-FRI-1120-KH_G1	Control/Aux/Rx Building	1.04	2.08E-05	1.22E-07	1.17E-07	4.83E-09
IE-FRI-1120-KH_G4	Control/Aux/Rx Building	1.04	2.06E-05	3.92E-07	3.76E-07	1.55E-08
IE-FRI-1120-KH_H1	Control/Aux/Rx Building	1.04	2.08E-05	1.11E-07	1.06E-07	4.38E-09
IE-FRI-1120-KH_H2	Control/Aux/Rx Building	1.04	2.06E-05	2.96E-07	2.85E-07	1.17E-08
IE-FRI-1120-KH_J4	Control/Aux/Rx Building	1.04	2.05E-05	1.25E-07	1.20E-07	4.93E-09
IE-FRI-1120-KH_J5	Control/Aux/Rx Building	1.04	2.57E-06	4.27E-08	4.10E-08	1.69E-09
IE-FRI-1120-KH_K3	Control/Aux/Rx Building	1.04	3.08E-05	3.14E-07	3.02E-07	1.24E-08
IE-FRI-1120-KH_K4	Control/Aux/Rx Building	1.04	2.94E-06	2.97E-08	2.86E-08	1.18E-09
IE-FRI-1120-KH_L5	Control/Aux/Rx Building	1.04	5.14E-06	7.52E-08	7.23E-08	2.98E-09
IE-FRI-1120-KH_M4	Control/Aux/Rx Building	1.04	8.45E-06	2.55E-07	2.45E-07	1.01E-08
IE-FRI-1120-KH_M5	Control/Aux/Rx Building	1.04	5.14E-06	2.19E-07	2.10E-07	8.66E-09
IE-FRI-1120-KH_TR04	Control/Aux/Rx Building	1.04	3.73E-07	1.45E-09	1.39E-09	5.73E-11
IE-FRI-1120-KH_TR09	Control/Aux/Rx Building	1.04	3.73E-07	6.18E-09	5.93E-09	2.44E-10
IE-FRI-1121-KG_B1	Control/Aux/Rx Building	1.04	1.24E-04	3.11E-07	2.98E-07	1.23E-08
IE-FRI-1121-KG_D1_RR	Control/Aux/Rx Building	1.04	3.74E-04	6.52E-08	6.26E-08	2.58E-09
IE-FRI-1121-KG_E1	Control/Aux/Rx Building	1.04	6.22E-05	9.60E-07	9.22E-07	3.80E-08
IE-FRI-1121-KG_TR01	Control/Aux/Rx Building	1.04	5.18E-07	1.87E-08	1.80E-08	7.42E-10
IE-FRI-1133B-KK_D2	Control/Aux/Rx Building	1.04	4.41E-08	1.03E-08	9.85E-09	4.06E-10
IE-FRI-1133B-KK_E0	Control/Aux/Rx Building	1.04	9.68E-05	3.37E-07	3.23E-07	1.33E-08
IE-FRI-1133B-KK_H0	Control/Aux/Rx Building	1.04	2.06E-05	6.47E-07	6.21E-07	2.56E-08
IE-FRI-1140A-S1_E	Containment	0.38	3.64E-06	3.85E-08	1.01E-07	-6.24E-08
IE-FRI-1140A-S1_I_RR	Containment	0.38	4.18E-04	6.90E-09	1.81E-08	-1.12E-08

Table 19-8 Delta CDF of Fire Ignition Frequency Sensitivity

L3-FPRA Fire Scenario	Generic Fire Location	Ratio[2]	Updated FIF[1]	Sensitivity CDF (/rcy)	Nominal CDF (/rcy)	Delta CDF
IE-FRI-1140B-S1_B	Containment	0.38	2.07E-04	3.55E-07	9.31E-07	-5.76E-07
IE-FRI-1140B-S1_E	Containment	0.38	3.28E-05	2.37E-07	6.22E-07	-3.85E-07
IE-FRI-1140B-S1_H	Containment	0.38	5.65E-05	2.24E-07	5.88E-07	-3.63E-07
IE-FRI-1140C-S1_C_RR	Containment	0.38	1.06E-03	1.18E-08	3.10E-08	-1.91E-08
IE-FRI-1140C-S1_L2	Containment	0.38	2.95E-04	5.67E-08	1.49E-07	-9.19E-08
IE-FRI-1144-T6_JB3_RR	Control/Aux/Rx Building	1.04	8.97E-05	6.26E-09	6.02E-09	2.48E-10
IE-FRI-1146-VF_TR01_RR	Plant Wide Buildings	1.92	5.30E-03	2.80E-06	1.46E-06	1.34E-06
IE-FRI-1149-DO_TR02_RR	Control/Aux/Rx Building	1.04	6.11E-07	2.89E-10	2.77E-10	1.14E-11
IE-FRI-1151-IQ_TR03_RR	Control/Aux/Rx Building	1.04	4.35E-06	4.36E-08	4.19E-08	1.72E-09
IE-FRI-1152-IN_TR01_RR	Control/Aux/Rx Building	1.04	1.47E-03	1.06E-08	1.02E-08	4.20E-10
IE-FRI-1152-IN_TR02	Control/Aux/Rx Building	1.04	5.24E-07	8.02E-09	7.70E-09	3.17E-10
IE-FRI-1153-IQ_TR01	Control/Aux/Rx Building	1.04	6.01E-05	9.94E-08	9.54E-08	3.93E-09
IE-FRI-1155-V1_TR01_RR	Control/Aux/Rx Building	1.04	7.42E-05	3.77E-08	3.62E-08	1.49E-09
IE-FRI-1156-V2_B	Control/Aux/Rx Building	1.04	1.00E-04	6.57E-08	6.31E-08	2.60E-09
IE-FRI-1157A-V3_B1	Control/Aux/Rx Building	1.04	2.04E-05	6.81E-07	6.54E-07	2.69E-08
IE-FRI-1160A-V8_C_RR	Plant Wide Buildings	1.92	1.49E-03	7.06E-07	3.67E-07	3.39E-07
IE-FRI-1161-T1_B	Diesel Generator Room	1.55	4.68E-03	7.26E-07	4.69E-07	2.57E-07
IE-FRI-1162-T2_B	Diesel Generator Room	1.55	4.68E-03	7.70E-07	4.97E-07	2.73E-07
IE-FRI-1163-T3_A_RR	Diesel Generator Room	1.55	4.28E-03	4.88E-07	3.15E-07	1.73E-07
IE-FRI-1173-JH_TR02	Control/Aux/Rx Building	1.04	5.27E-06	4.75E-08	4.56E-08	1.88E-09
IE-FRI-1174-JG_TR04_RR	Control/Aux/Rx Building	1.04	7.45E-05	1.74E-07	1.67E-07	6.89E-09
IE-FRI-1175-JI_TR01	Control/Aux/Rx Building	1.04	3.80E-05	6.79E-08	6.52E-08	2.69E-09
IE-FRI-1176-K1_TR04	Control/Aux/Rx Building	1.04	5.27E-06	3.40E-08	3.26E-08	1.34E-09
IE-FRI-1176-K1_TR05	Control/Aux/Rx Building	1.04	5.27E-06	2.43E-07	2.33E-07	9.61E-09
IE-FRI-1176-K1_TR06	Control/Aux/Rx Building	1.04	3.80E-08	2.03E-09	1.95E-09	8.04E-11
IE-FRI-1176-K1_TR07	Control/Aux/Rx Building	1.04	2.66E-06	3.18E-08	3.06E-08	1.26E-09
IE-FRI-1179-KV_B1_RR	Control/Aux/Rx Building	1.04	1.17E-04	8.00E-09	7.69E-09	3.17E-10
IE-FRI-1188-VH_A_RR	Plant Wide Buildings	1.92	1.26E-02	4.70E-07	2.44E-07	2.25E-07
IE-FRI-1300A-X1_A_RR	Plant Wide Buildings	1.92	1.48E-04	8.85E-10	4.60E-10	4.25E-10
IE-FRI-1503_TR01_RR	Battery Room	0.60	8.48E-05	5.66E-10	9.42E-10	-3.75E-10
IE-FRI-1506_B1	Turbine Building	1.05	4.76E-06	2.83E-07	2.71E-07	1.23E-08
IE-FRI-1506_JB1	Turbine Building	1.05	1.05E-04	3.46E-07	3.31E-07	1.50E-08
IE-FRI-1507_B1	Turbine Building	1.05	4.76E-06	3.64E-07	3.48E-07	1.57E-08
IE-FRI-1508_TR01	Turbine Building	1.05	4.08E-06	3.13E-07	3.00E-07	1.36E-08
IE-FRI-1509_Q_RR	Turbine Building	1.05	1.83E-02	1.94E-07	1.85E-07	8.39E-09
IE-FRI-1512_B0	Turbine Building	1.05	3.10E-05	5.06E-08	4.84E-08	2.19E-09
IE-FRI-1512_C0_RR	Turbine Building	1.05	1.54E-04	1.44E-07	1.37E-07	6.21E-09
IE-FRI-1512_C2	Turbine Building	1.05	1.36E-04	1.25E-07	1.19E-07	5.40E-09
IE-FRI-1512_D1	Turbine Building	1.05	6.54E-05	5.91E-08	5.66E-08	2.56E-09
IE-FRI-1512_K0	Turbine Building	1.05	3.10E-05	5.06E-08	4.84E-08	2.19E-09
IE-FRI-1512_K2	Turbine Building	1.05	2.63E-04	4.35E-07	4.16E-07	1.88E-08
IE-FRI-1530_A_RR	Plant Wide Buildings	1.92	1.01E-01	1.11E-06	5.76E-07	5.31E-07
IE-FRI-1603-KD_E_RR	Control/Aux/Rx Building	1.04	1.00E-02	1.05E-07	1.01E-07	4.14E-09
IE-FRI-1800_A_RR	Plant Wide Buildings	1.92	7.91E-02	8.46E-07	4.40E-07	4.06E-07
IE-FRI-2050-D4_A_RR	Control/Aux/Rx Building	1.04	3.42E-02	3.65E-07	3.50E-07	1.44E-08
IE-FRI-2080-M9_H1	Control/Aux/Rx Building	1.04	1.25E-03	1.20E-06	1.15E-06	4.74E-08
IE-FRI-2085-NB_TR04_RR	Control/Aux/Rx Building	1.04	8.14E-05	7.54E-08	7.25E-08	2.98E-09

Table 19-8 Delta CDF of Fire Ignition Frequency Sensitivity

L3-FPRA Fire Scenario	Generic Fire Location	Ratio[2]	Updated FIF[1]	Sensitivity CDF (/rcy)	Nominal CDF (/rcy)	Delta CDF
IE-FRI-2091-N4_B100	Control/Aux/Rx Building	1.04	3.55E-04	3.23E-07	3.10E-07	1.28E-08
IE-FRI-2095-N8_JB1	Control/Aux/Rx Building	1.04	1.37E-03	9.87E-09	9.48E-09	3.91E-10
IE-FRI-2098-N9_JB1_RR	Control/Aux/Rx Building	1.04	6.21E-04	4.37E-09	4.19E-09	1.73E-10
IE-FRI-2115-JZ_A_RR	Plant Wide Buildings	1.92	8.06E-02	8.63E-07	4.49E-07	4.14E-07
IE-FRI-2133A-KK_A_RR	Battery Room	0.60	7.34E-03	7.67E-08	1.28E-07	-5.09E-08
IE-FRI-2136-LP_A_RR	Control/Aux/Rx Building	1.04	3.49E-03	2.58E-08	2.48E-08	1.02E-09
IE-FRI-A040-BX_A_RR	Control/Aux/Rx Building	1.04	5.85E-03	1.53E-07	1.47E-07	6.06E-09
IE-FRI-A105-JY_ABN4	Control Room	5.96	8.35E-07	8.35E-07	1.40E-07	6.95E-07
IE-FRI-A105-JY_AI	Control Room	5.96	2.37E-04	7.60E-07	1.28E-07	6.32E-07
IE-FRI-A105-JY_AR	Control Room	5.96	4.74E-04	2.31E-06	3.88E-07	1.92E-06
IE-FRI-A105-JY_AT0	Control Room	5.96	1.66E-06	4.05E-08	6.81E-09	3.37E-08
IE-FRI-A105-JY_AT1	Control Room	5.96	5.93E-05	3.99E-07	6.70E-08	3.32E-07
IE-FRI-A105-JY_AT2	Control Room	5.96	5.93E-05	3.15E-07	5.29E-08	2.62E-07
IE-FRI-A105-JY_AT3	Control Room	5.96	5.93E-05	1.01E-06	1.70E-07	8.42E-07
IE-FRI-A105-JY_AV	Control Room	5.96	2.37E-04	1.15E-06	1.93E-07	9.57E-07
IE-FRI-A105-JY_AW0	Control Room	5.96	1.66E-06	3.85E-08	6.46E-09	3.20E-08
IE-FRI-A105-JY_AW1	Control Room	5.96	5.93E-05	3.98E-07	6.68E-08	3.31E-07
IE-FRI-A105-JY_AW2	Control Room	5.96	5.93E-05	3.12E-07	5.24E-08	2.60E-07
IE-FRI-A105-JY_AW3	Control Room	5.96	5.93E-05	9.35E-07	1.57E-07	7.78E-07
IE-FRI-A105-JY_AX	Control Room	5.96	2.37E-04	1.60E-05	2.68E-06	1.33E-05
IE-FRI-A105-JY_L	Control Room	5.96	4.74E-04	1.54E-06	2.58E-07	1.28E-06
IE-FRI-A105-JY_M	Control Room	5.96	2.37E-04	1.38E-06	2.32E-07	1.15E-06
IE-FRI-A105-JY_P2	Control Room	5.96	2.63E-05	1.24E-05	2.08E-06	1.03E-05
IE-FRI-A105-JY_Q1	Control Room	5.96	2.63E-05	1.16E-06	1.95E-07	9.66E-07
IE-FRI-A105-JY_Q6	Control Room	5.96	1.24E-05	8.58E-07	1.44E-07	7.14E-07
IE-FRI-A105-JY_R0	Control Room	5.96	1.24E-05	3.50E-07	5.87E-08	2.91E-07
IE-FRI-A105-JY_S2	Control Room	5.96	5.25E-05	3.12E-06	5.24E-07	2.60E-06
IE-FRI-A105-JY_S3	Control Room	5.96	2.63E-05	1.56E-06	2.62E-07	1.30E-06
IE-FRI-A105-JY_S5	Control Room	5.96	1.87E-05	1.11E-06	1.87E-07	9.26E-07
IE-FRI-A105-JY_U3A	Control Room	5.96	3.68E-07	9.73E-08	1.63E-08	8.10E-08
IE-FRI-A105-JY_U5A	Control Room	5.96	1.84E-07	1.08E-08	1.82E-09	9.00E-09
IE-FRI-A105-JY_U7A	Control Room	5.96	1.84E-07	6.60E-10	1.11E-10	5.49E-10
IE-FRI-AHVSWYD_B	Transformer Yard	2.27	3.61E-04	3.30E-07	1.45E-07	1.85E-07
IE-FRI-AHVSWYD_C	Transformer Yard	2.27	3.61E-04	3.30E-07	1.46E-07	1.85E-07
IE-FRI-AHVSWYD_E	Transformer Yard	2.27	1.27E-03	1.17E-06	5.17E-07	6.57E-07
IE-FRI-AHVSWYD_TR01	Transformer Yard	2.27	1.61E-04	1.49E-07	6.58E-08	8.36E-08
IE-FRI-ALVSWYD_TR01	Transformer Yard	2.27	3.52E-05	1.44E-07	6.34E-08	8.05E-08
IE-FRI-TB1_A	Plant Wide Buildings	1.92	3.69E-06	2.39E-07	1.24E-07	1.15E-07
IE-FRI-YARD_TR01	Plant Wide Buildings	1.92	1.39E-08	1.39E-08	7.24E-09	6.68E-09
IE-FRI-YARD_TR05	Plant Wide Buildings	1.92	1.54E-03	1.12E-08	5.84E-09	5.39E-09
IE-FRI-YARD_TR10	Plant Wide Buildings	1.92	4.96E-05	2.03E-07	1.06E-07	9.74E-08
				1.07E-04	6.14E-05	4.56E-05

Note:
1. RP-FPRA fire ignition frequency * Ratio
2. Average ratio of updated FIF to nominal FIF based on location

19.4.4 L3-FPRA Future Work

Performance of the sensitivity analyses described above, as well as other insights derived through the course of performing the L3-FPRA study, have led to the identification of some candidate areas for future work. A list of some of these areas is provided below.

- Apply the more recent fire ignition frequencies from NUREG-2169 if the necessary plant-specific information can be obtained to identify the specific ignition sources in each fire location.

- Expand the residual fire scenarios using the same mapping rules as were used for the fire sequences with higher CDF. This expansion would have little impact on the overall Level 1 fire CDF; however, as discussed under "RP-FPRA fire sequences" in Table 19-2, it might have a noticeable impact on some release category frequencies in the Level 2 PRA.

- Develop proper fire scenario settings within the fire scenario event tree, via placing basic events at each branch node (if possible). This would eliminate the requirement of performing an end state gather to reduce the individual sequence cut sets down to a single fire scenario cut set group. This would benefit the Level 2 analysis process. For the Level 2 analysis to ask the correct questions, it needs to understand the sequence path. Since the fire scenario cut sets are grouped together a lot of the sequence path information is lost and assumptions have to be made to perform the Level 2 analysis. This fire scenario event tree modification would slow the quantification time down and additional (manual) cut set removal would be required.

- Perform a more detailed HRA dependency analysis. The current HRA dependency analysis looked at just the top 200 HFE combination cut sets. If a larger number of cut sets were evaluated, a few more dependent HFEs could be identified.

- Perform a more detailed analysis of MCR abandonment scenarios, taking credit for control and operation of the plant from the remote shutdown panels. The detailed analysis should also implement the guidance documented in EPRI 3002013023 (EPRI, 2019).

- As discussed in Section 19.4.3.2, a more accurate modeling of MCR abandonment scenarios could reduce their CDF contribution by a factor of 10 or more. While MCR abandonment scenarios do not make a major contribution to total fire CDF, they do make a disproportionate contribution to LERF in the RP-FPRA.

- Enhance the level of fire PRA realism by incorporating the results of recent and ongoing research (performed by the NRC independently or in collaboration with EPRI) that advances understanding of fire phenomena and improves fire PRA methods, tools, and data. Specific examples include more realistic modeling of electrical cabinet fire growth and propagation and main control board fires, revised heat release rate distributions, and incorporation of the potential for manual fire detection and suppression prior to automatic detection. While it is currently unknown what ultimate effect these fire modeling improvements would have on the reported fire CDF for the L3-FPRA, the relatively high significance of electrical cabinet fires in the L3-FPRA results implies there could be substantial impact.

20 TASK 16 – FIRE PRA DOCUMENTATION

20.1 Objective of the Task

Task 16 in NUREG/CR-6850 provides a suggested outline for documenting the fire PRA along with supporting documentation items.

20.2 Reference Plant Work Performed on the Task

The RP-FPRA was documented by the reference plant in multiple reports. These reports were provided to the L3PRA project team.

20.3 SNL Review of the Reference Plant's Approach to Address the Task

SNL reviewed the RP-FPRA documentation and found the list of information was complete.

20.4 L3-FPRA Approach to Address the Task

This report and its appendices document the work performed to develop the L3-FPRA and its results.

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APPENDIX A
DEPENDENCY ANALYSIS FOR FIRE HUMAN FAILURE EVENTS

This appendix documents the dependency analysis for human failure events (HFEs). The factors considered in the dependency analysis are described in Section 16.4.2.2. From the 65 HFE dependency combinations, there were 59 unique HFE pairs to be evaluated for dependency. Analysis details are provided for each of the following HFE pairs:

A.1 CHG-XHE-NORMAL-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

A.2 OA-ALTAFW----H-FIRE and CHG-XHE-NORMAL-FIRE (Zero Dependence)

A.3 OAR_HPMSO----H-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

A.4 OAR_HPSLA----H-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

A.5 OAT----------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.6 OAT----------H-FIRE and OAC_NC-------H-FIRE (Moderate Dependence)

A.7 OAT----------H-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

A.8 OAT-ISINJ----H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.9 RCS-XHE-XM-TRIP-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.10 RCS-XHE-XM-TRIP-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

A.11 OAR_HPSLA----H-FIRE and OA-ALTAFW----H-FIRE (Zero Dependence)

A.12 OAR_HPMSO----H-FIRE and OA-ALTAFW----H-FIRE (Zero Dependence)

A.13 OAC_AC-------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.14 OAN_SL-------H-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

A.15 OAT-ISINJ----H-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

A.16 RCS-XHE-XM-TRIP-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

A.17 OACONTROL--AFW-FIRE and OAB_TR-------H-FIRE (Moderate Dependence)

A.18 OACONTROL--AFW-FIRE and OAR_LTFB-TRA-H-FIRE (Zero Dependence)

A.19 OAT-ISINJ----H-FIRE and OAC_NC-------H-FIRE (Moderate Dependence)

A.20 OA-ISOLETDOWNH-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

A.21 OAISOLSTMTDAFW-FIRE and OAR_LTFB-TRA-H-FIRE (Zero Dependence)

A.22 OAC_AC-------H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.23 OACONTROL--AFW-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.24 OA-START-AFW-H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.25 OA-START-AFW-H-FIRE and OAB_TR-------H-FIRE (High Dependence)

A.26 OAISOLSTMTDAFW-FIRE and OAB_TR-------H-FIRE (Moderate Dependence)

A.27 RCS-XHE-XM-TRIP-FIRE and OAC_NC-------H-FIRE (Zero Dependence)

A.28 OA-ISOLETDOWNH-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.29 OAT----------H-FIRE and OAR_LPSL-----H-FIRE (Low Dependence)

A.30 OAT----------H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.31 OAT----------H-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

A.32 OAT----------H-FIRE and OAC_AC-------H-FIRE (Moderate Dependence)

A.33 OACONTROL--AFW-FIRE and OA-CSISOL----H-FIRE (High Dependence)

A.34 OA-CSISOL----H-FIRE and OA-START-AFW-H-FIRE (High Dependence)

A.35 OA-ISOLETDOWNH-FIRE and OAC_AC-------H-FIRE (Low Dependence)

A.36 OAT-ISINJ----H-FIRE and OAC_AC-------H-FIRE (Moderate Dependence)

A.37 OA-CSISOL----H-FIRE and OAC_AC-------H-FIRE (Low Dependence)

A.38 RCS-XHE-XM-TRIP-FIRE and OAC_AC-------H-FIRE (Zero Dependence)

A.39 OAT-ISINJ----H-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

A.40 OAT-ISINJ----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.41 OA-ISOLETDOWNH-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

A.42 OACONTROL--AFW-FIRE and OAISOLSTMTDAFW-FIRE (Complete Dependence)

A.43 OAT----------H-FIRE and OA-CSISOL----H-FIRE (High Dependence)

A.44 OAT----------H-FIRE and OAISOLSTMTDAFW-FIRE (High Dependence)

A.45 OA-CSISOL----H-FIRE and OA-ISOLETDOWNH-FIRE (High Dependence)

A.46 RCS-XHE-XM-TRIP-FIRE and OA-ESFAS-HE1-H-FIRE (Zero Dependence)

A.47 OA-CSISOL----H-FIRE and OAT-ISINJ----H-FIRE (High Dependence)

A.48 OAR_HPSLA----H-FIRE and OAR_LPSL-----H-FIRE (Complete Dependence)

A.49 OA-OLP_SL----H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.50 OAR_HPMSO----H-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

A.51 OA-OLP_SL----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.52 OA-CSISOL----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.53 OA-START-AFW-H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.54 OAC_NC-------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

A.55 OAISOLSTMTDAFW-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

A.56 OA-ISOLETDOWNH-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

A.57 OA-ESFAS-HE1-H-FIRE and OAC_AC-------H-FIRE (Low Dependence)

A.58 OA-ISOLETDOWNH-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

A.59 OAISOLSTMTDAFW-FIRE and OAC_NC-------H-FIRE (Low Dependence)

A.1 CHG-XHE-NORMAL-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

1st HFE, CHG-XHE-NORMAL-FIRE, "Operator Fails to Establish Charging Given a Loss of RCP Seal Injection - Fire"

T_{delay} = 20 minutes

T_m = 2 minutes

Initial Cue(s): Loss of RCP Seal Inject or Charging Flow

Cognitive Procedure: Reactor Trip Response, step 10b

Execution Procedure: Chemical and Volume Control System

2nd HFE, CAD-XHE-SAFESTBLE-FIRE, "Operator Fails to Depressurize Secondary (72HR SAFE/STABLE) - Fire"

T_{delay} = 2400 minutes

Initial Cue(s): Core-Exit Thermocouple Temperature (711°F)

Cognitive Procedure: Response to Inadequate Cooling, step 15

Execution Procedure: Response to Inadequate Cooling, step 15

SAME CREW

There is greater than 12 hours between these HFEs and different crews are expected to perform each action; therefore, there is no dependency between these HFEs.

A.2 OA-ALTAFW----H-FIRE and CHG-XHE-NORMAL-FIRE (Zero Dependence)

1st HFE, OA-ALTAFW----H-FIRE, "Operator Fails to Provide Additional Water Source for Long Term AFW - Fire"

T_{delay} = 17.39 hours

T_m = 20 minutes

Initial Cue(s): CST level at less than 15%

Cognitive Procedure: Steam Generator Tube Rupture, foldout page

Execution Procedure: Auxiliary Feedwater System

2nd HFE, CHG-XHE-NORMAL-FIRE, "Operator Fails to Establish Charging Given a Loss of RCP Seal Injection - Fire"

T_{delay} = 20 minutes

Initial Cue(s): Loss of RCP Seal Inject or Charging Flow

Cognitive Procedure: Reactor Trip Response, step 10b

Execution Procedure: Chemical and Volume Control System

SAME CREW

There is greater than 12 hours between these HFEs and different crews are expected to perform each action; therefore, there is no dependency between these HFEs.

A.3 OAR_HPMSO----H-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

1st HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

T_m = 14 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, CAD-XHE-SAFESTBLE-FIRE, "Operator Fails to Depressurize Secondary (72HR SAFE/STABLE) - Fire"

T_{delay} = 2400 minutes

Initial Cue(s): Core-Exit Thermocouple Temperature (711°F)

Cognitive Procedure: Response to Inadequate Cooling, step 15

Execution Procedure: Response to Inadequate Cooling

SAME CREW

There is greater than 12 hours between these HFEs and different crews are expected to perform each action; therefore, there is no dependency between these HFEs.

A.4 OAR_HPSLA----H-FIRE and CAD-XHE-SAFESTBLE-FIRE (Zero Dependence)

1st HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

T_m = 11 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, CAD-XHE-SAFESTBLE-FIRE, "Operator Fails to Depressurize Secondary (72HR SAFE/STABLE) - Fire"

Tdelay = 2400 minutes

Initial Cue(s): Core-Exit Thermocouple Temperature (711°F)

Cognitive Procedure: Response to Inadequate Cooling, step 15

Execution Procedure: Response to Inadequate Cooling

SAME CREW

There is greater than 12 hours between these HFEs and different crews are expected to perform each action; therefore, there is no dependency between these HFEs.

A.5 OAT----------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.6 OAT----------H-FIRE and OAC_NC-------H-FIRE (Moderate Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

Tdelay = 20 minutes

Tm = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAC_NC-------H-FIRE, "Operator Fails to Initiate Normal Cooldown After LOCA with HPI - Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 20 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.7 OAT----------H-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

Initial Cue(s): RCS WR hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.8 OAT-ISINJ----H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.9 RCS-XHE-XM-TRIP-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.10 RCS-XHE-XM-TRIP-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization – SLOCA, RHR Failed, CCUs Available - Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST level decreased below 39%

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 23 or foldout page item 5

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.11 OAR_HPSLA----H-FIRE and OA-ALTAFW----H-FIRE (Zero Dependence)

1st HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

T_m = 11 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, OA-ALTAFW----H-FIRE, "Operator Fails to Provide Additional Water Source for Long Term AFW - Fire"

T_{delay} = 17.39 hours

Initial Cue(s): CST level at less than 15%

Cognitive Procedure: Steam Generator Tube Rupture, foldout page

Execution Procedure: Auxiliary Feedwater System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.12 OAR_HPMSO----H-FIRE and OA-ALTAFW----H-FIRE (Zero Dependence)

1st HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

T_m = 14 minute

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, OA-ALTAFW----H-FIRE, "Operator Fails to Provide Additional Water Source for Long Term AFW - Fire"

T_{delay} = 17.39 hours

Initial Cue(s): CST level at less than 15%

Cognitive Procedure: Steam Generator Tube Rupture, foldout page

Execution Procedure: Auxiliary Feedwater System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.13 OAC_AC-------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

T_m = 30 seconds

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"[19]

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.14 OAN_SL-------H-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

1st HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

T_m = 19 minutes

Initial Cue(s): RCS WR hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

[19] In this scenario, HPR is queried because HPI has not actually failed. Spurious operation of the containment spray system has drained the RWST, so if operators fail to depressurize for LPR, they can still attempt HPR.

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization – SLOCA, RHR Failed, CCUs Available - Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST level decreased below 39%

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 23 or foldout page item 5

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.15 OAT-ISINJ----H-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

Initial Cue(s): RCS WR hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.16 RCS-XHE-XM-TRIP-FIRE and OAN SL-------H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

Initial Cue(s): RCS WR hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.17 OACONTROL--AFW-FIRE and OAB_TR-------H-FIRE (Moderate Dependence)

1st HFE, OACONTROL--AFW-FIRE, "Operator Fails to Control AFW Flow Given Spurious – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 13 a/b

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAB_TR-------H-FIRE, "Operator Fails to Feed and Bleed - Transient - Fire"

T_{delay} = 24 minutes

Initial Cue(s): WR Level in any 3 SGs – Less than 29% [44% adverse] -OR- RCS Pressure – Greater than 2335 psig due to loss of Secondary Heat Sink

Cognitive Procedure: Response to Loss of Secondary Heat Sink, step 6

Execution Procedure: Response to Loss of Secondary Heat Sink

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 17 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.18 OACONTROL--AFW-FIRE and OAR_LTFB-TRA-H-FIRE (Zero Dependence)

1st HFE, OACONTROL--AFW-FIRE, "Operator Fails to Control AFW Flow Given Spurious – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 13 a/b

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_LTFB-TRA-H-FIRE, "Operator Fails to Establish HPR for Long Term F&B - Fire"

T_{delay} = 577.8 minutes

Initial Cue(s): RWST level lowers to less than 29%

Cognitive Procedure: Response to Loss of Secondary Heat Sink, step 48

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.19 OAT-ISINJ----H-FIRE and OAC_NC-------H-FIRE (Moderate Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAC_NC-------H-FIRE, "Operator Fails to Initiate Normal Cooldown After LOCA with HPI - Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 20 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.20 OA-ISOLETDOWNH-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization – SLOCA, RHR Failed, CCUs Available - Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST level decreased below 39%

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 23 or foldout page item 5

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be about an hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.21 OAISOLSTMTDAFW-FIRE and OAR_LTFB-TRA-H-FIRE (Zero Dependence)

1st HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer), Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_LTFB-TRA-H-FIRE, "Operator Fails to Establish HPR for Long Term F&B - Fire"

T_{delay} = 577.8 minutes

Initial Cue(s): RWST level lowers to less than 29%

Cognitive Procedure: Response to Loss of Secondary Heat Sink, step 48

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.22 OAC_AC-------H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

T_m = 30 seconds

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"[20]

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.23 OACONTROL--AFW-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OACONTROL--AFW-FIRE, "Operator Fails to Control AFW Flow Given Spurious – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 13 a/b

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

Tdelay = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

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20 In this scenario, HPR is queried because HPI has not actually failed. Spurious operation of the containment spray system has drained the RWST, so if operators fail to depressurize for LPR, they can still attempt HPR.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.24 OA-START-AFW-H-FIRE and OAR HPMSO----H-FIRE (Zero Dependence)

1st HFE, OA-START-AFW-H-FIRE, "Operator Fails to Manually Start AFW Pumps in MCR – Fire"

T_{delay} = 16 minutes

T_m = 0 minutes

Initial Cue(s): Total feed flow capability to SGs – not greater than 570 GPM available

Cognitive Procedure: Reactor Trip Response, step 7

Execution Procedure: Reactor Trip Response

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.25 OA-START-AFW-H-FIRE and OAB_TR-------H-FIRE (High Dependence)

1st HFE, OA-START-AFW-H-FIRE, "Operator Fails to Manually Start AFW Pumps in MCR – Fire"

T_{delay} = 16 minutes

T_m = 0 minutes

Initial Cue(s): Total feed flow capability to SGs – not greater than 570 GPM available

Cognitive Procedure: Reactor Trip Response, step 7

Execution Procedure: Reactor Trip Response

2nd HFE, OAB_TR-------H-FIRE, "Operator Fails to Feed and Bleed - Transient - Fire"

T_{delay} = 24 minutes

Initial Cue(s): WR Level in any 3 SGs – Less than 29% [44% adverse] -OR- RCS Pressure – Greater than 2335 psig due to loss of Secondary Heat Sink

Cognitive Procedure: Response to Loss of Secondary Heat Sink, step 6

Execution Procedure: Response to Loss of Secondary Heat Sink

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 8 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.26 OAISOLSTMTDAFW-FIRE and OAB_TR-------H-FIRE (Moderate Dependence)

1st HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAB_TR-------H-FIRE, "Operator Fails to Feed and Bleed - Transient - Fire"

T_{delay} = 24 minutes

Initial Cue(s): WR Level in any 3 SGs – Less than 29% [44% adverse] -OR- RCS Pressure – Greater than 2335 psig due to loss of Secondary Heat Sink

Cognitive Procedure: Response to Loss of Secondary Heat Sink, step 6

Execution Procedure: Response to Loss of Secondary Heat Sink

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 17 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.27 RCS-XHE-XM-TRIP-FIRE and OAC_NC-------H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OAC_NC-------H-FIRE, "Operator Fails to Initiate Normal Cooldown After LOCA with HPI - Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 30 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs. However, as discussed earlier in Section 16.4.2.2, given the special nature of the potential dependencies involving RCS-XHE-XM-TRIP-FIRE, zero dependency was assumed between these HFEs.

A.28 OA-ISOLETDOWNH-FIRE and OAR HPSLA----H-FIRE (Zero Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

Tdelay = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.29 OAT----------H-FIRE and OAR_LPSL-----H-FIRE (Low Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization – SLOCA, RHR Failed, CCUs Available – Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST Level Decreased Below 29%

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 15 or Foldout page, Item 6

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 44 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is low dependency between these HFEs.

A.30 OAT----------H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR – RWST MSO – Fire"

T_{delay} = 684.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of Secondary Heat Sink, step 48

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.31 OAT----------H-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OA-OLP_SL----H-FIRE, "Operator Fails to Restart RHR Pump for LPI SLOCA HPI Fails DPI Success – Fire"

T_{delay} = 129 minutes

Initial Cue(s): core exit thermocouples ≥ 711 °F; RCS SUBCOOLING ≤ 24 °F; NO RCPs running; RVLIS full range ≤ 48%

Cognitive Procedure: Critical Safety Function Status Tree, F-0.2

Execution Procedure: Respond to Degraded Core Cooling

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.32 OAT----------H-FIRE and OAC_AC-------H-FIRE (Moderate Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 20 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.33 OACONTROL--AFW-FIRE and OA-CSISOL----H-FIRE (High Dependence)

1st HFE, OACONTROL--AFW-FIRE, "Operator Fails to Control AFW Flow Given Spurious – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 13 a/b

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 8 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.34 OA-CSISOL----H-FIRE and OA-START-AFW-H-FIRE (High Dependence)

1st HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

T_m = 1 minute

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

2nd HFE, OA-START-AFW-H-FIRE, "Operator Fails to Manually Start AFW Pumps in MCR – Fire"

T_{delay} = 16 minutes

Initial Cue(s): Total feed flow capability to SGs – not greater than 570 GPM available

Cognitive Procedure: Reactor Trip Response, step 7

Execution Procedure: Reactor Trip Response

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 1 minute.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.35 OA-ISOLETDOWNH-FIRE and OAC_AC-------H-FIRE (Low Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 35 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is low dependency between these HFEs.

A.36 OAT-ISINJ----H-FIRE and OAC_AC-------H-FIRE (Moderate Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 20 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs.

A.37 OA-CSISOL----H-FIRE and OAC_AC-------H-FIRE (Low Dependence)

1st HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

T_m = 1 minute

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

<u>SAME CREW</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

<u>COMMON COGNITIVE</u>

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

<u>SAME TIME/TIMING</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 25 minutes.

<u>SAME LOCATION</u>

Yes, both actions are performed in the control room; therefore, there is low dependency between these HFEs.

A.38 RCS-XHE-XM-TRIP-FIRE and OAC AC-------H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

<u>SAME CREW</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 30 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is moderate dependency between these HFEs. However, as discussed earlier in Section 16.4.2.2, given the special nature of the potential dependencies involving RCS-XHE-XM-TRIP-FIRE, zero dependency was assumed between these HFEs.

A.39 OAT-ISINJ----H-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OA-OLP_SL----H-FIRE, "Operator Fails to Restart RHR Pump for LPI SLOCA HPI Fails DPI Success – Fire"

T_{delay} = 129 minutes

Initial Cue(s): core exit thermocouples≥ 711 °F; RCS subcooling ≤ 24 °F; NO RCPs running; RVLIS full range ≤ 48%

Cognitive Procedure: Critical Safety Function Status Tree, F-0.2

Execution Procedure: Response to Degraded Core Cooling

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.40 OAT-ISINJ----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR – RWST MSO – Fire"

T_{delay} = 684.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of Secondary Heat Sink, step 48

Execution Procedure: Transfer to Cold Leg Recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.41 OA-ISOLETDOWNH-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

Initial Cue(s): RCS WR hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.42 OACONTROL--AFW-FIRE and OAISOLSTMTDAFW-FIRE (Complete Dependence)

1st HFE, OACONTROL--AFW-FIRE, "Operator Fails to Control AFW Flow Given Spurious – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 13 a/b

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have the same cue and procedures; therefore, they do have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, these actions are expected to occur at the same time.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is complete dependency between these HFEs.

A.43 OAT----------H-FIRE and OA-CSISOL----H-FIRE (High Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 5 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.44 OAT----------H-FIRE and OAISOLSTMTDAFW-FIRE (High Dependence)

1st HFE, OAT----------H-FIRE, "Operator Fails to Terminate SI – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

2nd HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 13 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.45 OA-CSISOL----H-FIRE and OA-ISOLETDOWNH-FIRE (High Dependence)

1st HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

T_m = 1 minute

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

2nd HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

Tdelay = 5 minutes

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 10 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.46 RCS-XHE-XM-TRIP-FIRE and OA-ESFAS-HE1-H-FIRE (Zero Dependence)

1st HFE, RCS-XHE-XM-TRIP-FIRE, "Operator Fails to Trip Reactor Coolant Pumps"

T_{delay} = 10 minutes

T_m = 10 seconds

Initial Cue(s): Loss of ACCW

Cognitive Procedure: Reactor Trip or Safety Injection, OATC initial action 11

Execution Procedure: Reactor Trip or Safety Injection, OATC initial action 11

2nd HFE, OA-ESFAS-HE1-H-FIRE, "Operator Fails to Start Equipment on Failure of ESFAS Signal – Fire"

T_{delay} = 8.5 minutes

Initial Cue(s): SI is activated

Cognitive Procedure: Reactor Trip or Safety Injection, step 7

Execution Procedure: Reactor Trip or Safety Injection

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 2 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs. However, as discussed earlier in Section 16.4.2.2, given the special nature of the potential dependencies involving RCS-XHE-XM-TRIP-FIRE, zero dependency was assumed between these HFEs.

A.47 OA-CSISOL----H-FIRE and OAT-ISINJ----H-FIRE (High Dependence)

1st HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

T_m = 1 minute

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

2nd HFE, OAT-ISINJ----H-FIRE, "Operator Fails to Terminate SI After ISINJ Initiating Event – Fire"

T_{delay} = 20 minutes

T_m = 9 minutes

Initial Cue(s): RCS Subcooling – greater than 24°F [38°F adverse]

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 11

Execution Procedure: SI Termination

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 5 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is high dependency between these HFEs.

A.48 OAR_HPSLA----H-FIRE and OAR_LPSL-----H-FIRE (Complete Dependence)

1st HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

T_m = 11 minute

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization per ES-1.2 – SLOCA, RHR Failed, CCUs Available – Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST Level Decreased Below 29%

Cognitive Procedure: Loss of reactor or secondary coolant, step 15 or Foldout page, Item 6

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have the same cue and similar procedures; therefore, they do have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is complete dependency between these HFEs.

A.49 OA-OLP_SL----H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, OA-OLP_SL-----H-FIRE, "Operator Fails to Restart RHR Pump for LPI SLOCA HPI Fails DPI Success – Fire"

T_{delay} = 129 minutes

T_m = 5 minutes

Initial Cue(s): core exit thermocouples ≥ 711 °F; RCS subcooling ≤ 24 °F; NO RCPs running; RVLIS full range ≤ 48%

Cognitive Procedure: Critical Safety Function Status Tree, F-0.2

Execution Procedure: Respond to Degraded Core Cooling

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs – Fire"[21]

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

<u>SAME CREW</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

<u>COMMON COGNITIVE</u>

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

<u>SAME TIME/TIMING</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

<u>SAME LOCATION</u>

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.50 OAR_HPMSO----H-FIRE and OAR_LPSL-----H-FIRE (Zero Dependence)

1st HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

T_m = 14 minute

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

2nd HFE, OAR_LPSL-----H-FIRE, "Operator Fails to Establish LPR After Depressurization – SLOCA, RHR Failed, CCUs Available – Fire"

T_{delay} = 64 minutes

Initial Cue(s): RWST Level Decreased Below 29%

Cognitive Procedure: Loss of reactor or secondary coolant, step 15 or Foldout page, Item 6

Execution Procedure: Transfer to cold leg recirculation

[21] In this scenario, HPR is queried because HPI has not actually failed. Spurious operation of the containment spray system has drained the RWST, so if operators fail to depressurize for LPR, they can still attempt HPR.

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.51 OA-OLP_SL----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OA-OLP_SL----H-FIRE, "Operator Fails to Restart RHR Pump for LPI SLOCA HPI Fails DPI Success – Fire"

T_{delay} = 129 minutes

T_m = 5 minutes

Initial Cue(s): core exit thermocouples ≥ 711 °F; RCS subcooling ≤ 24 °F; NO RCPs running; RVLIS full range ≤ 48%

Cognitive Procedure: Critical Safety Function Status Tree, F-0.2

Execution Procedure: Respond to Degraded Core Cooling

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.52 OA-CSISOL----H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OA-CSISOL----H-FIRE, "Operator Fails to Close CS Suction From the RWST – Fire"

T_{delay} = 15 minutes

T_m = 1 minute

Initial Cue(s): RWST level reduction

Cognitive Procedure: Loss of Reactor or Secondary Coolant, Foldout page item 5

Execution Procedure: Loss of Reactor or Secondary Coolant

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.53 OA-START-AFW-H-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OA-START-AFW-H-FIRE, "Operator Fails to Manually Start AFW Pumps in MCR – Fire"

T_{delay} = 16 minutes

T_m = 0 minutes

Initial Cue(s): Total feed flow capability to SGs – not greater than 570 GPM available

Cognitive Procedure: Reactor Trip Response, step 7

Execution Procedure: Reactor Trip Response

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

<u>SAME CREW</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

<u>COMMON COGNITIVE</u>

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

<u>SAME TIME/TIMING</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

<u>SAME LOCATION</u>

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.54 OAC_NC-------H-FIRE and OAR_HPSLA----H-FIRE (Zero Dependence)

1st HFE, OAC_NC-------H-FIRE, "Operator Fails to Initiate Normal Cooldown After LOCA with HPI - Fire"

T_{delay} = 40 minutes

T_m = 0.5 seconds

Initial Cue(s): RCS pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

2nd HFE, OAR_HPSLA----H-FIRE, "Operator Fails to Establish HPR - SLOCA with CCUs - Fire"

T_{delay} = 368.4 minutes

Initial Cue(s): RWST level < 29%

Cognitive Procedure: Loss of reactor or secondary coolant, foldout page (item 5)

Execution Procedure: Transfer to cold leg recirculation

<u>SAME CREW</u>

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures (or steps); therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.55 OAISOLSTMTDAFW-FIRE and OAN_SL-------H-FIRE (Zero Dependence)

1st HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAN_SL-------H-FIRE, "Operator Fails to Establish Normal RHR – SLOCA - Fire"

T_{delay} = 280 minutes

Initial Cue(s): RCS Wide Range hot leg temperatures – less than 350°F; RCS pressure – less than 350 psig

Cognitive Procedure: Post-LOCA Cooldown and Depressurization, step 46

Execution Procedure: Residual Heat Removal System

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.56 OA-ISOLETDOWNH-FIRE and OA-OLP_SL----H-FIRE (Zero Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, “Operator Fails to Isolate Letdown Upstream of RV – Fire”

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OA-OLP_SL----H-FIRE, “Operator Fails to Restart RHR Pump for LPI SLOCA HPI Fails DPI Success – Fire”

T_{delay} = 129 minutes

Initial Cue(s): core exit thermocouples ≥ 711 °F; RCS subcooling ≤ 24 °F; NO RCPs running; RVLIS full range ≤ 48%

Cognitive Procedure: Critical Safety Function Status Tree, F-0.2

Execution Procedure: Respond to Degraded Core Cooling

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.57 OA-ESFAS-HE1-H-FIRE and OAC_AC-------H-FIRE (Low Dependence)

1st HFE, OA-ESFAS-HE1-H-FIRE, “Operator Fails to Start Equipment on Failure of ESFAS Signal – Fire”

T_{delay} = 8.5 minutes

T_m = 9 minutes

Initial Cue(s): SI is activated

Cognitive Procedure: Reactor Trip or Safety Injection, step 7

Execution Procedure: Reactor Trip or Safety Injection

2nd HFE, OAC_AC-------H-FIRE, "Operator Fails to Depressurize for LPI – SLOCA HPI Failed – Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS Pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 32 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is low dependency between these HFEs.

A.58 OA-ISOLETDOWNH-FIRE and OAR_HPMSO----H-FIRE (Zero Dependence)

1st HFE, OA-ISOLETDOWNH-FIRE, "Operator Fails to Isolate Letdown Upstream of RV – Fire"

T_{delay} = 5 minutes

T_m = 1 minute

Initial Cue(s): Action 3 in Annunciator Response Procedure for Fire Alarm Computer

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, step 3.a

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAR_HPMSO----H-FIRE, "Operator Fails to Establish HPR - RWST MSO - Fire"

T_{delay} = 670.8 minutes

Initial Cue(s): Initial HPR alignment attempt failed

Cognitive Procedure: Loss of secondary heatsink, step 48

Execution Procedure: Transfer to cold leg recirculation

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be greater than 1 hour.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is zero dependency between these HFEs.

A.59 OAISOLSTMTDAFW-FIRE and OAC_NC-------H-FIRE (Low Dependence)

1st HFE, OAISOLSTMTDAFW-FIRE, "Operator Fails to Isolate Steam to the TD-AFW Pump – Fire"

T_{delay} = 7 minutes

T_m = 2 minutes

Initial Cue(s): S/G level high

Cognitive Procedure: Annunciator Response Procedure for Fire Alarm Computer, Table 3, step 14

Execution Procedure: Annunciator Response Procedure for Fire Alarm Computer

2nd HFE, OAC_NC-------H-FIRE, "Operator Fails to Initiate Normal Cooldown After LOCA (ES-1.2) with HPI - Fire"

T_{delay} = 40 minutes

Initial Cue(s): RCS pressure – Greater than 300 psig

Cognitive Procedure: Loss of Reactor or Secondary Coolant, step 22

Execution Procedure: Post-LOCA Cooldown and Depressurization

SAME CREW

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be less than 12 hours. Therefore, the crew is expected to be the same.

COMMON COGNITIVE

The actions have diverse cues and different procedures; therefore, they do not have common cognitive function.

SAME TIME/TIMING

Precise timings for these HFEs are not available; however, the time between HFEs is expected to be on the order of 33 minutes.

SAME LOCATION

Yes, both actions are performed in the control room; therefore, there is low dependency between these HFEs.

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Alan Kuritzky, NRC Level 3 PRA Project Program Manager

11. ABSTRACT (200 words or less)
The U.S. Nuclear Regulatory Commission performed a full-scope site Level 3 probabilistic risk assessment (PRA) project (L3PRA project) for a two-unit pressurized-water reactor reference plant. The scope of the L3PRA project encompasses all major radiological sources on the site (i.e., reactors, spent fuel pools, and dry cask storage), all internal and external hazards, and all modes of plant operation. A full-scope site Level 3 PRA for a nuclear power plant site can provide valuable insights into the importance of various risk contributors by assessing accidents involving one or more reactor cores as well as other site radiological sources. This report, one of a series of reports documenting the models and analyses supporting the L3PRA project, specifically addresses the reactor, at-power, Level 1 PRA model for internal fires for a single unit. The analyses documented herein are based information for the reference plant as it was designed and operated as of 2012 and does not reflect the plant as it is currently designed, licensed, operated, or maintained.

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