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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Integrated Site Risk, All Hazards, Level 1, Level 2, and Level 3 PRA

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U.S. NRC Level 3 Probabilistic Risk Assessment Project

Integrated Site Risk, All Hazards, Level 1, Level 2, and Level 3 PRA

Manuscript Completed: May 2025
Date Published: August 2025

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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 integrated site risk task, which encompasses risk contributions from all major radiological sources on the reference site (i.e., reactors, spent fuel pools, and dry cask storage). The analyses documented herein are based on information for the reference plant as it was designed and operated as of 2012 and do not reflect the plant as it is currently designed, licensed, operated, or maintained.[1]

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, 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] To provide results and insights better aligned with the current design and operation of the reference plant, this report also provides the results of a parametric sensitivity analysis based on a set of new plant equipment and PRA model assumptions for all three PRA levels (referred to as the "FLEX sensitivity case"). The sensitivity analysis reflects the current reactor coolant pump shutdown seal design at the reference plant, as well as 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 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. 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.**

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

[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.

- 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.

For the integrated site risk task, specifically, the staff has gained important experience related to multi-unit PRA and sitewide risk calculations. Multi-unit PRA, in particular, has become important for the U.S. and some countries with either existing or planned multiple units on site.

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.

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. As such, this report will not be the sole basis for any regulatory decisions specific to the reference plant.

This report provides the approach and results for the integrated site risk (ISR) task that supports the L3PRA project. ISR, which includes all major radiological sources on site (i.e., reactors, spent fuel pools [SFPs], and dry cask storage [DCS]), has not been included in the scope of previous NRC Level 3 PRA studies, such as NUREG-1150 (ML040140729). However, SECY-11-0089 specifically identifies all major on-site radiological sources as being within the scope of this study.

[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.

ISR Terminology

This report provides the approach and results for the integrated site risk (ISR) task that supports the L3PRA project. ISR has no formal definition within the PRA community. For the purposes of the L3PRA project, the following definition has been adopted:

> Integrated site risk is the total combined risk of a release from one or more radiological sources on site (i.e., reactors, spent fuel pools, and dry cask storage), considering all hazards and all plant operating states.

If single source (i.e., single reactor, spent fuel pool [SFP], and dry cask storage [DCS] facility) PRAs were performed for all hazards and operating states, a simplistic approach for estimating ISR, or total risk, would be to assume that all the radiological sources are independent and, therefore, simply sum the results from all these PRAs. However, a significant limitation of this approach is that it overlooks that there are dependencies between radiological sources (e.g., there are common initiating events that can cause all reactors on site to trip). In addition, this approach does not provide any useful PRA insights beyond what the single source PRAs have already provided.

The impact of cross-unit dependencies is illustrated in Figure ES-1. This figure shows a Venn diagram for core damage frequency for two, identical reactors on a site. In this Venn diagram, the following terms are used:

- Single-unit (SU) CDF that is calculated by traditional PRAs (i.e., SUCDF)
- Multi-unit CDF (MUCDF), which is the area of overlap that represents both units experiencing core damage simultaneously
- Single-unit only CDF (SOCDF), which represents only one unit experiencing core damage

Figure ES-1 also illustrates why adding together the CDF results for the two reactors overestimates sitewide CDF for a two-unit site. Instead, using Figure ES-1, Level 1 sitewide CDF (SCDF) for a site containing only two reactors is calculated as:

$$SCDF = SUCDF_1 + SUCDF_2 - MUCDF$$

Note that Figure ES-1 shows that MUCDF will always be some fraction of the SUCDF (i.e., MUCDF cannot be larger than SUCDF). This is true for total SUCDF (i.e., sum of CDF for all types of initiators) as well as for individual initiators that have been identified as cross-unit or multi-unit initiators (MUIEs). The single unit only CDF (SOCDF) is represented by the area outside of the overlapping circles (i.e., accident sequences that involve only one unit at a time experiencing core damage).

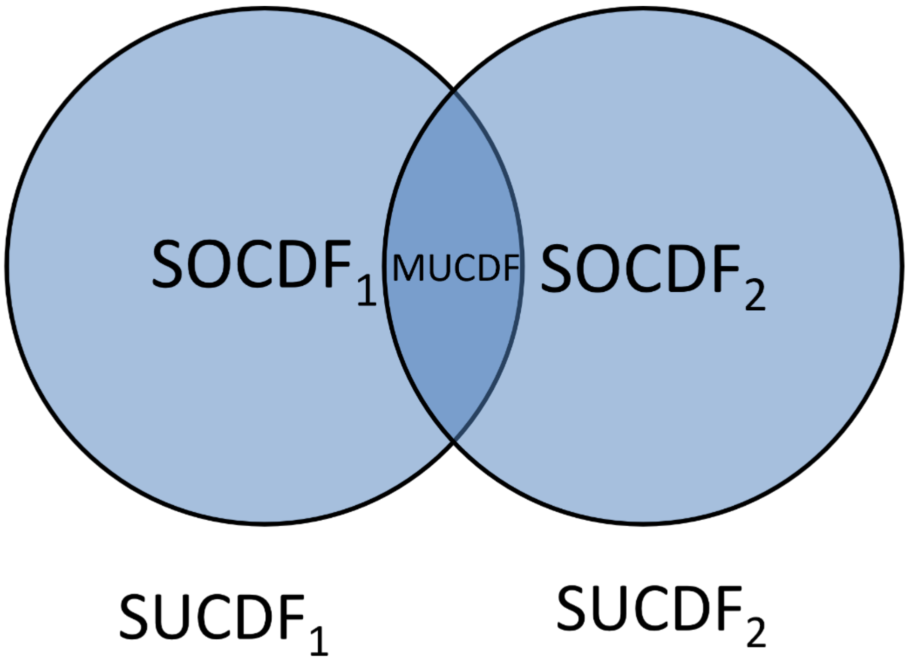

Figure ES-1 Venn Diagram Showing Single Unit and Multi-Unit CDF for Two Units

The fraction of SUCDF that represents MUCDF (i.e., the area of overlap) will vary by the strength of the dependencies between the two reactors. For example, Figure ES-1 illustrates the case in which the fraction of SUCDF that represents MUCDF is expected to be relatively small. On the other hand, Figure ES-2 illustrates the case in which the fraction of SUCDF that represents MUCDF is expected to be much larger (e.g., if the two reactors share many systems and/or have other significant dependencies).

Similar in concept to Figure ES-1 and Figure ES-2, the L3PRA project's ISR task is focused on the "overlap" between the two reactors and the SFP (i.e., accidents involving both reactors and the SFP). As for Figure ES-1 and Figure ES-2, the estimation of ISR for the two reactors and the SFP would be overestimated if the single source risks were simply added (i.e., dependencies between the radiological sources were ignored).

The equation above and the figures above demonstrate that the key to estimating total or integrated risk is to estimate the risk for the areas of overlap. For the ISR task, these areas of overlap are identified as MU risk and multi-source risk, respectively, and defined as follows:

- MU risk is the risk from concurrent or simultaneous accidents for multiple reactors (i.e., both reactors on the two-unit reference site).
- Multi-source risk is the risk from concurrent or simultaneous accidents for multiple reactors (i.e., both reactors on the reference site) and another radiological source (i.e., either the SFPs or the DCS on the reference site).

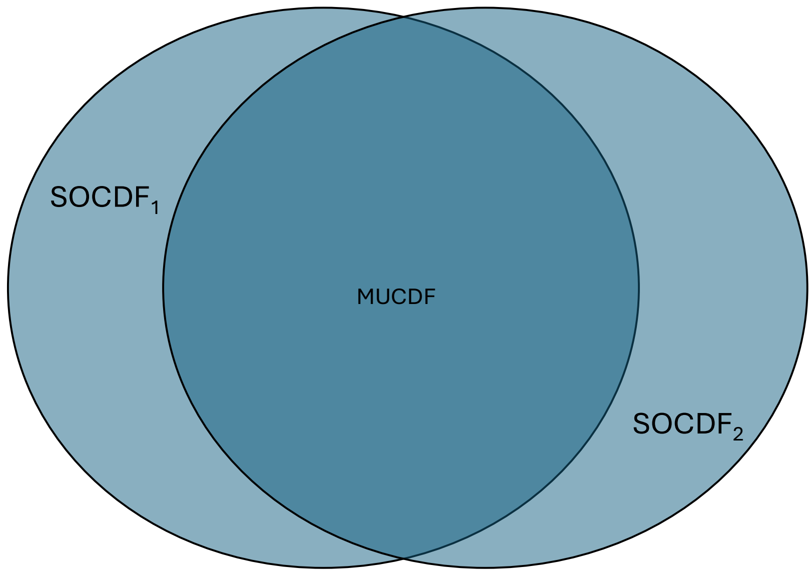

Figure ES-2 Venn Diagram Showing Single Unit and Multi-Unit CDF for Two Units That Have Many Cross-Unit Dependencies

Note that the timing of the accidents is key to the definitions above. For example, a multi-source risk scenario for two reactors and an SFP starts with an initiating event that affects all radiological sources and is followed by accident sequences in which there are dependencies between the reactors (i.e., multi-unit or cross-unit) and dependencies between the SFP and the reactors. MU risk and multi-source risk also represent the most challenging scenarios for accident response.

Because the definitions for MU and multi-source risk focus on certain very challenging accidents, there is an opportunity for PRA results to provide additional insights and to identify potential vulnerabilities that the single source PRA cannot provide. However, well-known computational challenges for MU Level 2 PRA limited the ISR task to consideration of only one or two accident scenarios. As a result, while MUCDF was calculated for all relevant hazards, MU Level 2 and MU Level 3 results were developed only for illustrative scenarios. For these reasons, the Level 3 PRA project's ISR task focused on estimating MU and multi-source risk only. In other words, no specific calculations for ISR were performed.[2]

ISR Technical Approach

The ISR task was performed principally using the results of the L3PRA project's PRAs and underlying plant-specific information that are documented in other project reports. The expertise

[2] In principle, ISR results could be developed for the illustrative scenarios at the individual initiator level (e.g., for the seismic bin 6 initiating event). However, the ISR task did not develop all the necessary inputs for this calculation. In particular, accident scenarios that involved the SFPs and a single reactor were not addressed by the Level 3 PRA project's ISR task. In addition, as noted above, such results would not provide any insights that the single source PRAs do not already provide.

and experience of the L3PRA project team also were important to the development of the ISR approach and its implementation.

The steps to perform the ISR task are:

- Step 1: Specify scope, screening criteria, and risk metrics
- Step 2: Perform sitewide dependency assessment
- Step 3: Perform multi-unit (MU) (and sitewide) initiating events analysis
- Step 4: Estimate MU Level 1 core damage frequency (CDF)
- Step 5: Identify and define an illustrative multi-source scenario
- Step 6: Develop and quantify MU Level 2 PRA
- Step 7: Develop and quantify MU Level 3 PRA
- Step 8: Combine MU risk with risk from SFPs and DCS
- Step 9: Identify key sources of uncertainty

Note that, consistent with discussions provided above, the Level 3 PRA project's ISR task is focused on calculating MU and multi-source risk; ISR, per the definitions provided above, is not calculated. Calculation of ISR would require additional complex steps.

For the Level 3 PRA project, Step 4 was first performed for the at-power, internal events Level 1 PRA, then repeated for each hazard group (both internal and external). MUCDF results were generated by using single-unit PRA (SUPRA) cutsets and a cutset estimation method that was developed by the L3PRA project ISR team.

In recognition of limited project resources and limited computational ability for Level 2 PRA, Step 5 was added to identify and define a scenario in which both reactors and other relevant radiological sources (e.g., the SFPs) required accident mitigation. The results of Step 5 were used to focus Steps 6 and 7 (i.e., MU Level 2 and Level 3 PRA) on only two MU scenarios. Also, the results of Step 5 were used to focus Step 8 on a single illustrative multi-source scenario for which MU risks were integrated with those from the other major onsite radiological sources (i.e., the SFPs and DCS). As such, the L3PRA project ISR task serves primarily as a proof-of-concept study.

Sitewide Dependency Assessment

An important step in calculating MU and multi-source risk is a sitewide dependency assessment. As part of this step for the L3PRA project, multiple categories of cross-unit and sitewide dependencies (e.g., common initiating events; shared physical resources; shared systems, structures and components; identical components) were identified and used to calculate MU and multi-source risk.

Principal findings from the sitewide dependency assessment include the following:

- The two reactors on the reference site are mostly independent (i.e., "loosely coupled"), except for some identical components and shared hazards for external events.

- A subset of the initiating events addressed in the single unit PRAs are relevant to MU risk.

- Seismic events are the only initiating events that are important to both the reactors and the SFPs.
- There are very few cross-unit dependencies involving operator actions, except for operator actions in the reactor Level 2 PRA model.
- There is some resource sharing between the SFPs and the reactors for seismic events.
- There are no dependencies identified between the DCS facility and the two reactors and the SFPs.
- The sitewide dependencies that have been identified provide important insights by themselves.

MU Core Damage Frequency

The approach selected for calculating MUCDF in the L3PRA project has been labeled the "cutset estimation method" (CEM). Development of this approach was based on a thorough understanding of the plant-specific, potential dependencies between the two reactors on the reference site, the cutsets from the SUPRAs, and the potential impact on MU risk calculations from dependencies between the two reactors. The approach is based on a detailed review of the highest contributing SUPRA cutsets (typically comprising 95 percent or more of CDF) and applying "coupling factors" to account for the inter-unit dependencies (primarily, common-cause failures [CCFs] and seismic hazard correlations). A scaling factor is ultimately applied to account for the residual MUCDF associated with the unanalyzed cutsets.

At-power MUCDF was calculated for all identified multi-unit initiating events (MUIEs) with the following examples of contributors to total at-power MUCDF:

- The contribution to total MUCDF from all loss of offsite power (LOOP) events combined is about 14 percent, with grid-related LOOPs contributing the most (about 8 percent of total MUCDF).
- The contribution to total MUCDF from the loss of service water initiating event is about 25 percent.
- Seismic events (i.e., the sum of contributions from all eight bins modeled) contribute over half of the total MUCDF (about 53 percent) with:
    - Bins 3 through 6 contributing nearly 94 percent of the total seismic MUCDF.
    - Bins 4, 5 and 6 contributing over 80 percent of the total seismic MUCDF, in nearly equal shares.

Additional insights from the MUCDF results include:

- Only certain initiating events affect both reactors simultaneously (e.g., LOOPs, loss of service water, and external hazards).
- The two reactors on the reference site were assessed to be very independent of each other. This independence is reflected in the MUCDF results, for example:

- Total at-power MUCDF is only 10 percent of total, at-power single-unit CDF (SUCDF).
- The sum of at-power MUCDF for all four types of LOOP is less than 5 percent of the sum of at-power SUCDF for LOOPs.
- The at-power, wind MUCDF is less than 1 percent of at-power, wind SUCDF.

- However, for seismic bins 5, 6, 7, and 8 (the highest seismic hazard bins), MUCDF is 90 to 100 percent of SUCDF.
- MU coupling factors (i.e., factors that represent cross-unit dependencies computationally) play a very important role in the calculated MU risks:
    - For weather-related and grid-related LOOPs, cutsets containing MU CCF coupling factors account for 86 percent of MUCDF.
    - For seismic bin 2, cutsets containing seismic hazards correlations (e.g., MU coupling factors) account for about 64 percent of MUCDF.
    - For seismic bin 6, cutsets containing MU seismic hazard correlations account for 97 percent of MUCDF.

MU and Multi-Source Risk

Observations and insights for the overall ISR task are provided below—first for MU risk and then for multi-source (i.e., combined MU and SFP) risk.

To gain insight into multi-source (i.e., combined MU and SFP) risk, the L3PRA project team identified a unique multi-source initiating scenario that involves nearly simultaneous (i.e., within the traditional 24-hour mission time) consequences at both reactors and SFPs. At the reference plant, the two reactors and the two hydraulically connected SFPs were assessed to be independent, except for:

- Seismic bins 5 and 6 are important contributors to both MUCDF and the SFP Level 1 and 2 PRA results.
- Implementation of mitigation strategies for the reactors and SFPs share physical resources (e.g., portable pumps and water tanks) and associated human resources (e.g., operators).

Therefore, a multi-source scenario was defined for a seismic bin 6 event that:

- involves the sitewide dependencies mentioned above
- can be mitigated but requires mitigation for SFPs within 24 hours, unlike most of the accident scenarios addressed by the L3PRA SFP PRA, where mitigation is not required for many hours after accident initiation
- involves a timing dependency (i.e., given the amount of time it takes to complete the mitigative actions, they can be considered to occur in the similar timeframe):

        - the reactors require mitigation strategies to be completed by about 22 hours after reactor trip

        - the SFPs require mitigation strategies to be completed by about 10 hours after reactor trip

    - involves releases from both reactors and the SFPs

A summary of risk measures—single source (i.e., both single unit and SFP), MU, and multi-source—is provided in Table ES-1. While the values presented in Table ES-1 can be used to make general comparisons between single source, MU, and multi-source risk results, caution should be used in this comparison because the MU Level 3 PRA results do not include the full set of initiators analyzed in the single unit analyses. For example, other seismic bins are significant contributors to MUCDF and, therefore, also would be contributors to MU risk along with seismic bin 6 results. However, multi-source results for seismic bin 6 are expected to represent a significant fraction of multi-source risk because sitewide damage is less consequential for lower seismic bins and the frequency of occurrence is much lower for higher seismic bins.

The following are some general insights obtained from Table ES-1:

- The single unit, low power and shutdown (LPSD) risk for internal events only is similar to that for the single unit, at-power, for internal events (and internal floods).

- The all-hazards, all modes (i.e., both at power and LPSD) risk results for the SFPs are roughly one or two orders of magnitude smaller than the corresponding (summed) single unit risk results.

- MU risk results for the two representative MUIEs are roughly one or two orders of magnitude smaller than the SU results.

- The results for multi-source risk for seismic bin 6 are substantially smaller than the MU risk for seismic bin 6.

Table ES-1 Summary of Single Source, MU, and Multi-Source Risk Measures

Scope Piece	Release Freq. (/yr)	Individual Latent Fatality Risk, 0–10 mi (/yr)[1]	Collective Total Effective Dose Risk (person-rem/yr), 0–50 miles[1]	Total Economic Cost Risk, 0–50 mi (2015$/yr)[1]
At-power (internal events and internal floods) for a single unit	6.9E-05	2.5E-08	9.9	80,000
At-power (all hazards[2]) for a single unit	1.6E-04	6.4E-08	27	230,000
LPSD (internal events) for a single unit	1.2E-05	4.5E-09	3.6	72,000

SFP (all hazards)	5.8E-07	6.3E-10	0.52	8,400
All MU LOOPWR[3]	5.52E-07	3.35E-10	0.14	1,220
All MU EQK-BIN-6[4]	2.47E-06	1.84E-09	0.94	9,630
Multi-source (i.e., simultaneous accidents for both reactors and SFP) for seismic bin 6	3.10E-08	5.55E-11	0.041	589

Note 1: Results are a frequency-weighted sum of all RCs. Reactor at-power results are based on reactor-critical-years. To convert the risk metric of the **reactor at-power** from per reactor-critical year to per calendar-year, multiply the result by the plant availability factor of 0.93. Note, the resulting reactor, at-power, calendar-year risk does not include the risk associated with reactor shutdown operations during the calendar year.

Note 2: Includes at-power internal events, internal floods, internal fires, seismic events, and high-wind events. Also, note that these values (and the other values reported in this table) do not include credit for the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies (FLEX), as well as other more recent plant changes, such as the installation of new reactor coolant pump shutdown seals. When these other changes are credited, the total release frequency from a single unit is reduced to 1.0E-04/rcy.

Note 3: LOOPWR – weather-related LOOP; the results shown are for simultaneous core damage at both units due to dependencies such as shared, connected, or identical structures, systems, and components [SSCs]).

Note 4: EQK-BIN-6 – seismic hazard bin 6; the results shown are for simultaneous core damage at both units due to their co-location on the same site during a seismic bin 6 earthquake.

Additional Observations and Insights

There are several unique aspects of the L3PRA project's ISR analysis, including:

- To date, this is the only analysis that examined the combined risk of two reactors and two SFPs.
- A simplified approach was used to estimate MUCDF (which was justified, in part, by the lack of dependencies between the two reactors).
- Due to the lack of certain dependencies (e.g., shared SSCs) between the reactors and the SFPs, MU and SFP radiological releases could be added to serve as multi-source inputs for Level 3 PRA.
- As shown in the L3PRA single-unit (SU) Level 3 PRAs, it appears that source terms can be added to produce reasonable, yet conservative results as compared to those generated by the MACCS multi-source capability.

The L3PRA project's ISR task confirmed several state-of-the-art limitations, such as:

- If there are multiple reactors on site (i.e., more than two), the analysis of MU risk will get complicated very quickly. There will likely be numerous cross-combinations of cutsets and both the cutset estimation method (used in this study) and the more traditional event tree/fault tree approach may no longer be practical.

- Because there is limited data to support estimation of MU CCF coupling factors, conservative, generic coupling factors were used. As a result, the MUCDF estimates are expected to be conservative.

- Because there is little basis for MU seismic coupling factors, coupling factors were assigned using the SU seismic correlations and the expertise of the NRC's seismic PRA expert. Consequently, some of the MU seismic risk results may be conservative.

- As found in similar analyses, the L3PRA project's MU Level 2 PRA efforts were computationally challenging. Even for just two reactors, the number of MU release categories that could be addressed was limited.

- Like other similar analyses of combined reactor-SFP risk (e.g., EPRI 3002002691, "PWR Spent Fuel Pool Risk Assessment Integration Framework and Pilot Plant Application"), the L3PRA project's ISR task addressed only at-power conditions. However, the single-source LPSD PRAs for the reactor and SFPs show that LPSD risk is significant.

ACKNOWLEDGMENTS

The authors of this report would like to thank Latonia Enos-Sylla for her technical review of this report and Jonathan Evans and Kevin Coyne for their management review and guidance. The authors would also like to thank the members of the Level 3 PRA project technical advisory group for providing helpful comments on the draft report.

ABBREVIATIONS AND ACRONYMS

AC	alternating current
ACC	accumulator
ACCW	auxiliary component cooling water
ACRS	Advisory Committee on Reactor Safeguards
ADAMS	Agencywide Documents Access and Management System
AFW	auxiliary feedwater
AMSAC	anticipated transient without scram mitigation system actuation circuitry
ANS	American Nuclear Society
AOP	abnormal operating procedure
ARV	atmospheric relief valve
ASEP	Accident Sequence Evaluation Program
ASME	American Society of Mechanical Engineers
ATD	atmospheric transport and dispersion
ATWS	anticipated transient without scram
bar-abs	bars absolute pressure
BDD	Binary Decision Diagram
BMT	basemat melt-through
BE	basic event
CAFTA	Computer Aided Fault Tree Analysis tool for PRA
CBDT	Cause-Based Decision Tree
CCCG	common cause component group
CCF	common-cause failure
CCP	centrifugal charging pump
CCU	containment cooling unit
CCW	component cooling water
CDF	core damage frequency
CEM	cutset estimation method
CF	coupling factor (especially to represent multi-unit dependencies)
CFDP	conditional fuel damage probability for spent fuel pools
CIF	containment isolation failure
CSFST	critical safety function status tree
CST	condensate storage tank
CVCS	chemical and volume control system
DC	direct current
DCS	dry cask storage
DWST	demineralized water storage tank
EB	Empirical Bayes
ECCS	emergency core cooling system
ECWS	essential chilled water system
EDG	emergency diesel generator
EDMG	extensive damage mitigation guidance
EF	error factor
EOP	emergency operating procedure
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ESF	engineered safety features
ESFAS	engineered safety features actuation system
FHB	fuel handling building

FIP	Final Integrated Plan for implementation of FLEX strategies
FLEX	flexible and diverse mitigation strategies
FMCUB	Factored Minimal Cut Set Upper Bound
FMEA	failure mode and effects analysis
FTLR	fail to load/run
FTR	fail to run
FTS	fail to start
FV	Fussell-Vesely
FWST	fire water storage tank
GE	general emergency
gpm	gallons per minute
HCR	Human Cognitive Reliability
HEC	Hazardous Environmental Conditions
HEP	human error probability
HFE	human failure event
HLR	hot-leg recirculation
HPI	high-pressure injection
HPR	high-pressure recirculation
HRA	human reliability analysis
HVAC	heating, ventilation and air conditioning
IA	instrument air
IAEA	International Atomic Energy Agency
IE	initiating event; also “SUIE”
IEF	(single unit) initiating event frequency per reactor-critical-year (as typically calculated by Level 1 PRA for a single unit); also “SUIEF”
IEEE	Institute of Electrical and Electronic Engineers
INPO	Institute of Nuclear Power Operations
ICES	INPO Consolidated Events Database
ICF-BURN	intermediate containment failure due to burn
ISFSI	independent spent fuel storage installation
ISLOCA	interfacing system loss-of-coolant accident
ISR	integrated site risk
JCNRM	Joint Committee on Nuclear Risk Management
kV	kilovolt
L3PRA	Level 3 probabilistic risk assessment (project)
LERF	large early release frequency
LCF	late containment failure
LOINV	loss of inventory accident (for spent fuel pools)
LLOCA	large loss-of-coolant accident
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPI	low-pressure injection
LRF	large release frequency
LPR	low-pressure recirculation
LPSD	low power and shutdown
m	meter
m^3	cubic meters
MACCS	MELCOR Accident Consequence Code System
MCUB	minimal cutset upper bound
MELCOR	Method for Estimation of Leakages and Consequences of Releases
MDP	motor-driven pump

MOV	motor-operated valve
MCC	motor control center
MFIV	main feedwater isolation valves
MFW	main feedwater
MLOCA	medium loss-of-coolant accident
MSIV	main steam isolation valve
MSPI	mitigating systems performance index
MU	multi-unit
MUCDF	multi-unit core damage frequency per reactor-critical-year of an accident involving core damage to two or more reactor units on a multi-unit site (also "MU CDF")
MUIE	multi-unit initiating event for two or more reactors on the site (also, could also be a "sitewide IE")
MUIEF	multi-unit initiating event frequency per reactor-critical-year for two or more reactors on the site (also, could also be a "sitewide" initiating event frequency)
MUPRA	multi-unit PRA for two or more reactors on a multi-unit site
MURCF	multi-unit release category frequency
NCP	normal charging pump
NIS	nuclear instrumentation system
NPSH	net positive suction head
NRC	Nuclear Regulatory Commission
NSCW	nuclear service cooling water
OCP	operating cycle phase
ORE	Operator Reliability Experiments
PAU	physical analysis unit (fire area or compartment modeled in the fire PRA) (used synonymously with fire zone)
PGA	peak ground acceleration
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSF	performance shaping factor
psi	pounds per square inch
PWR	pressurized water reactor
RAT	reserve auxiliary transformer
RADS	Reliability and Availability Data System
RAW	risk achievement worth
RC	release category
RCF	release category frequency
RCP	reactor coolant pump
RCS	reactor coolant system
RCY	reactor-critical year
RHR	residual heat removal
RIR	Risk Increase Ratio
RMWST	reactor makeup water storage tank
ROP	Reactor Oversight Process
RPS	reactor protection system
RTB	reactor trip breaker
RWST	refueling water storage tank
SAMG	Severe Accident Guideline
SAPHIRE	Systems Analysis Programs for Hands-on Integrated Reliability Evaluations
SCUBE	SAPHIRE Cutset Upper Bound Estimator

SBO	station blackout
SFU	significant fuel uncovery in spent fuel pool
SFUF	significant fuel uncovery frequency in spent fuel pool (analogous to core damage frequency for reactors)
SFP	spent fuel pool
SFPCPS	spent fuel pool cooling and purification system
SFPS	Spent Fuel Pool Study (NUREG-2161)
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SLERF	site large early release frequency per reactor-critical-year of an accident involving a large early release, either from a single unit or from the combination of releases from multiple units.
SLOINV	small loss of inventory accident (for spent fuel pools)
SLOCA	small loss-of-coolant accident
SOC	site operating configuration
SOCDF	single unit <u>only</u> core damage frequency (i.e., PRA results for <u>only single unit contributions</u> on a multiple unit site)
SORV	stuck-open relief valve
SPAR	standardized plant analysis risk
SPRA	seismic PRA
SPT	standard penetration test
SRM	staff requirements memorandum
SRV	safety relief valve
SSB	secondary side break
SSC	structure, system, and component
SSIE	support system initiating event
SSPS	solid state protection system
SSU	safety system unavailability
SU	single unit
SUCDF	single unit core damage frequency in units of “per reactor-critical-year”—this is traditionally or typically calculated by Level 1 PRA for a single unit (i.e., “CDF”)
SUIE	single unit initiating event (as typically used for Level 1 PRA for a single unit)
SUIEF	single unit initiating event frequency in units of “per reactor-critical-year” (as typically used for Level 1 PRA for a single unit)
SUPRA	single unit PRA (as traditionally performed)
TAF	top of active fuel
TBV	turbine bypass valve
TDP	turbine-driven pump
THERP	Technique for Human Error Rate Prediction
TPCCW	turbine plant closed cooling water
TPCW	turbine plant cooling water system
TS	technical specifications
U1CDF	core damage frequency for Unit 1 calculated from a traditional Level 1 PRA (also single unit CDF)
UAT	unit auxiliary transformer
UET	unfavorable exposure time
V	volt
VCT	volume control tank
WOG	Westinghouse Owner’s Group

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 to:

- 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 provides the approach and results for the integrated site risk (ISR) task that supports the L3PRA project. Integrated site risk, which includes all major radiological sources on site (i.e., reactors, spent fuel pools, and dry cask storage), has not been included in the scope of previous NRC Level 3 PRA studies, such as NUREG-1150 (NRC, 1990). However, SECY-11-0089 specifically identifies all major on-site radiological sources as being within the scope of this study.

Licensee information used in 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,

[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 and multi-source 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.

assumptions, practices, methods, and conventions used by the NRC in the development of plant-specific PRA models. **As such, the L3PRA project reports will not be the sole basis for any regulatory decisions specific to the reference plant.**

To provide results and insights better aligned with the current design and operation of the reference plant, this report also provides the results of a parametric sensitivity analysis based on a set of new plant equipment and PRA model assumptions for all three PRA levels (referred to as the “FLEX sensitivity case”). 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, both of which reduce the risk to the public.

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 (no 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

CAUTION: The L3PRA project was developed to meet the specific objectives outlined in SECY-11-0089 using state of practice methods and data. While the study provides valuable insights and addresses its key objectives, 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.

1.1 Background on Integrated Site Risk and Multi-Unit PRA

The original scope of the all-hazards, sitewide L3PRA project included ISR (see NRC, 2011b and NRC, 2012a), which is intended to address all radiological sources on a nuclear power plant site. For the reference site, the following major radiological sources are included in the scope of the L3PRA project:

- Two (nearly identical) operating reactor units (Unit 1 and Unit 2)
- Two, hydraulically connected spent fuel pools (SFPs), one for each operating reactor unit (Unit 1 and Unit 2)
- An independent spent fuel storage installation (ISFSI), also referred to as a dry cask storage (DCS) facility

Previous NRC PRA efforts, such as NUREG-1150 (NRC, 1990), did not include consideration of risk from the SFPs or DCS, or risk of multi-unit accidents; therefore, ISR is a new PRA task. Furthermore, the project team is not aware of any previous ISR efforts. For example, Google and ScienceDirect search results for “integrated site risk” consist of mostly multi-unit PRA (MUPRA) efforts rather than multi-source risk estimates (which include all types of radiological sources). Also, the ISR team reached out to PRA experts who have worked (or are currently working) on MUPRAs, have developed MUPRA guidance, or are developing the MUPRA Standard. The results of this outreach confirmed that no ISR work is currently being performed for the existing fleet of reactors.

Although results from all radiological sources must be integrated for the ISR task, MU risk was considered first for the two nuclear power plants (NPPs) on the reference site. Then, the results for the SFPs and DCS were integrated with the MUPRA results.

Although PRA is traditionally performed for a single unit only, the Seabrook PRA (Pickard, Lowe, & Garrick, Inc. [PL&G], 1983) was an early industry PRA effort that considered two-unit risk.[3] In response to the 2011 Great Japan Earthquake and the events at the Fukushima Daiichi nuclear power station,[4] there has been increased interest in multi-unit risk in both the U.S. and internationally. There have been multiple responses to this event from the NRC, including

[3] Note that the second unit at the Seabrook site was never built.

[4] See, for example, the Government of Japan (2011, 2012), IAEA (2011), Institute of Nuclear Power Operations (INPO) (2011, 2012).

lessons learned (NRC, 2011c) and implications for PRA (Siu, 2013). Zhou (2021) summarizes the history and current status of MUPRA development.

The only comprehensive MUPRA efforts have been performed for Level 1 PRA. Efforts to perform MUPRA for Level 2 and Level 3 PRA have been limited because of the many release category (source term) combinations that would need to be evaluated. Simplified approaches or illustrative example scenarios have been addressed for Level 2 and Level 3 PRAs instead. Bixler and Kim (2021) describe this problem and their simplified approach.

In addition to the limited experience in performing MUPRAs, there is also limited guidance for developing MUPRAs. Three reports that provide such guidance were used extensively in the L3PRA project's ISR task:

1. International Atomic Energy Agency (IAEA), 2019, "Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units," Safety Report Series No. 96.

2. IAEA, 2021a, "Multi-Unit Probabilistic Safety Assessment, Safety Series Report 110.

3. Electric Power Research Institute (EPRI), 2021a, "Framework for Assessing Multi-Unit Risk to Support Risk-Informed Decision-Making - Phase 1 and 2: General Framework and Application-Specific Refinements," EPRI 3002020765.

It also should be noted that the American Nuclear Society (ANS)/American Society of Mechanical Engineers (ASME) Joint Committee on Nuclear Risk Management (JCNRM) is currently developing a MUPRA Standard for at-power, existing light water reactors (LWRs).

1.2 Definition of Integrated Site Risk

ISR has no formal definition within the PRA community. For the purposes of the L3PRA project, the following definition has been adopted:

> Integrated site risk is the total combined risk of a release from one or more radiological sources on a site (i.e., reactors, spent fuel pools, and dry cask storage), considering all hazards and all plant operating states.

If single source (i.e., single reactor, SFP, and DCS) PRAs were performed for all hazards and operating states, a simplistic approach for estimating ISR, or total risk, would be to assume that all the radiological sources are independent and, therefore, simply sum the results from all the PRAs. However, a significant limitation of this approach is that it overlooks that there are dependencies between radiological sources (e.g., there are common initiating events that can cause all reactors on site to trip). In addition, this approach does not provide any useful PRA insights beyond what the single source PRAs have already provided.

The impact of cross-unit dependencies is illustrated in Figure 1-1. This figure shows a Venn diagram for core damage frequency (CDF) for two identical reactors on a site. In this Venn diagram, the following terms are used:

- Single-unit (SU) CDF that is calculated by traditional PRAs (i.e., SUCDF)

- Multi-unit CDF (MUCDF), which is the area of overlap that represents both units experiencing core damage simultaneously

- Single-unit only CDF (SOCDF), which represents only one unit experiencing core damage

Figure 1-1 also illustrates why adding together the CDF results for the two reactors overestimates sitewide CDF for a two-unit site. Instead, using Figure 1-1, Level 1 sitewide CDF (SCDF) for a site containing only two reactors is calculated as:

$$SCDF = SUCDF_1 + SUCDF_2 - MUCDF$$

Note that Figure 1-1 shows that MUCDF will always be some fraction of SUCDF (i.e., MUCDF cannot be larger than SUCDF). This is true for total SUCDF (i.e., sum of CDF for all types of initiators) as well as for individual initiators that have been identified as cross-unit or multi-unit initiators (MUIEs). The single unit only CDF (SOCDF) is represented by the area outside of the overlapping circles (i.e., accident sequences that involve only one unit at a time experiencing core damage).

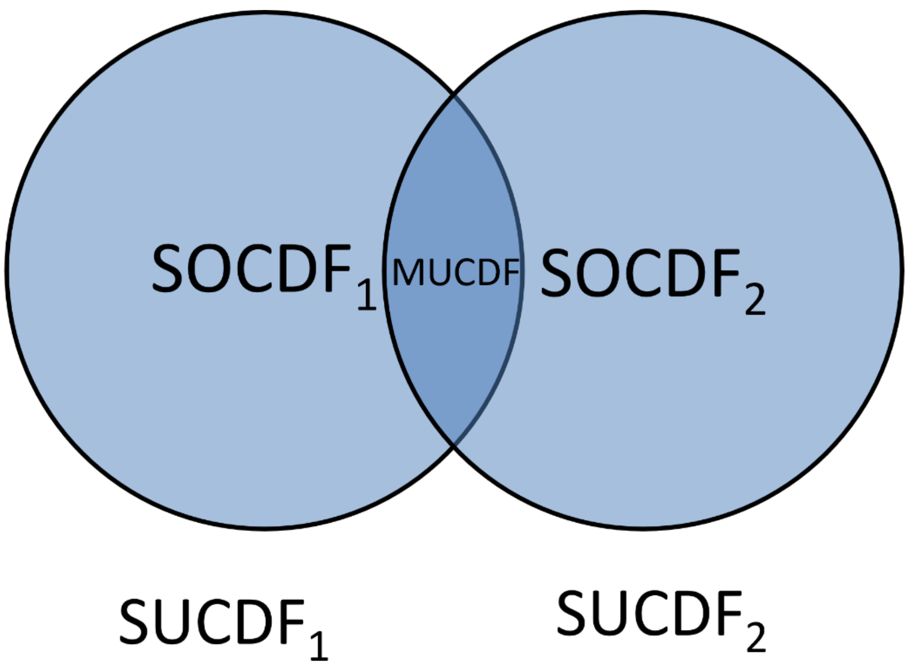

Figure 1-1 Venn Diagram Showing Single Unit and Multi-Unit Risk for Two Units

The fraction of SUCDF that represents MUCDF (i.e., the area of overlap) will vary by the strength of the dependencies between the two reactors. For example, Figure 1-1 illustrates the case in which the fraction of SUCDF that represents MUCDF is expected to be relatively small. On the other hand, Figure 1-2 illustrates the case in which the fraction of SUCDF that represents MUCDF is expected to be much larger (e.g., if the two reactors share many systems and/or have other significant dependencies). In contrast, Figure 1-3 shows the case for a single initiating event that is not a multi-unit initiating event, so there is no overlap between Unit 1 and Unit 2 CDF.

Similar in concept to Figure 1-1 and Figure 1-2, the L3PRA project's ISR task is focused on the "overlap" between the two reactors and the SFP (i.e., accidents involving both reactors and the SFP). As for Figure 1-1 and Figure 1-2, the estimation of ISR for the two reactors and the SFP would be overestimated if the single source risks were simply added (i.e., dependencies between the radiological sources were ignored).

The equation and figures above demonstrate that the key to estimating total or integrated risk is to estimate the risk for the areas of overlap. For the ISR task, these areas of overlap are identified as MU risk and multi-source risk, respectively, and defined as follows:

- MU risk is the risk from concurrent or simultaneous accidents for multiple reactors (i.e., both reactors on the two-unit reference site).
- Multi-source risk is the risk from concurrent or simultaneous accidents for multiple reactors (i.e., both reactors on the reference site) and another radiological source (i.e., either the SFPs or the DCS on the reference site).

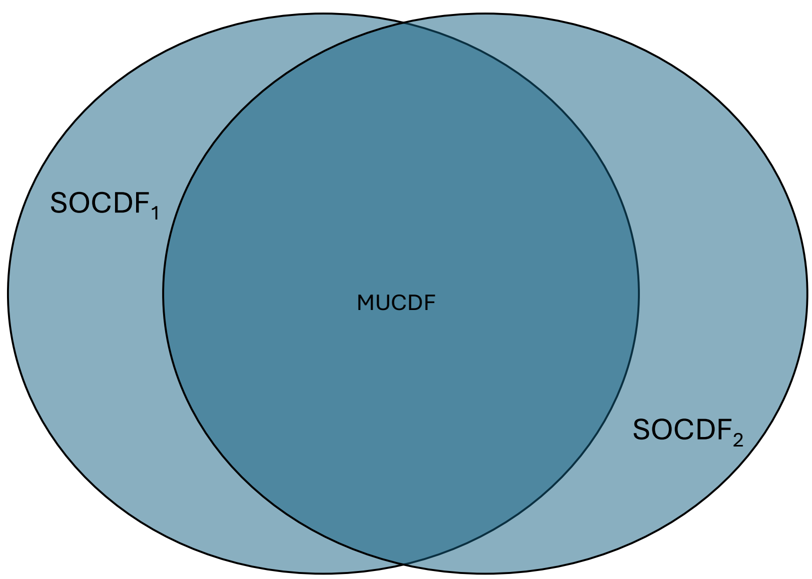

Figure 1-2 Venn Diagram Showing Single Unit and Multi-Unit CDF for Two Units That Have Many Cross-Unit Dependencies

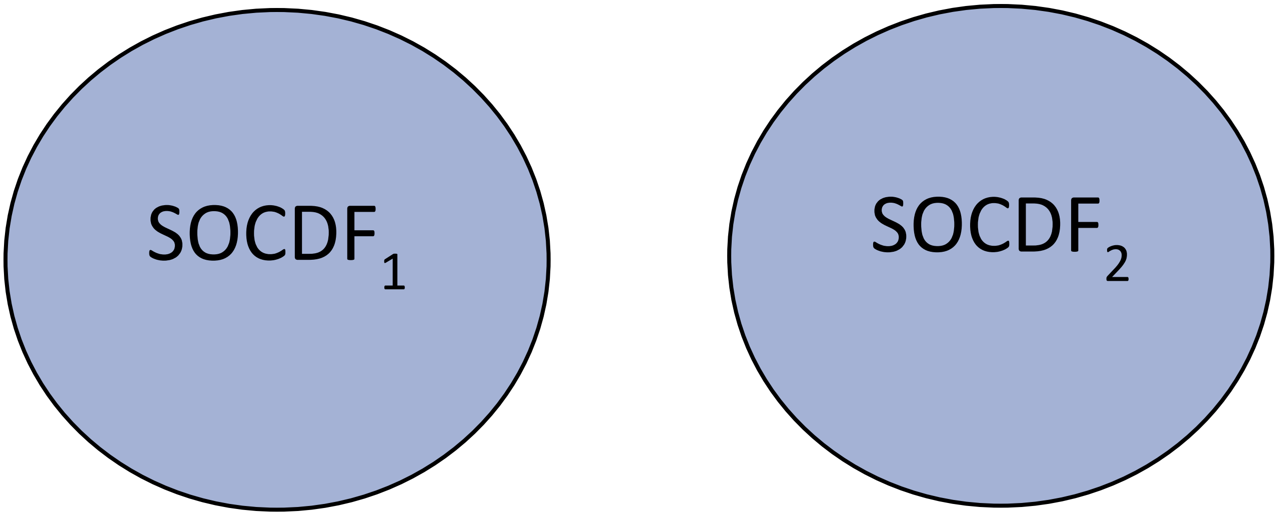

Figure 1-3 Venn Diagram Showing CDF for Two Units That Have No Cross-Unit Dependencies

Note that the timing of the accidents is key to the definitions above. For example, a multi-source risk scenario for two reactors and an SFP starts with an initiating event that affects all radiological sources and is followed by accident sequences in which there are dependencies between the reactors (i.e., multi-unit or cross-unit) and dependencies between the SFP and the reactors. MU risk and multi-source risk also represent the most challenging scenarios for accident response.

Because the definitions for MU and multi-source risk focus on certain very challenging accidents, there is an opportunity for PRA results to provide additional insights and to identify potential vulnerabilities that the single source PRA cannot provide. However, well-known computational challenges for MU Level 2 PRA limited the ISR task to consideration of only one or two accident scenarios. As a result, while MUCDF was calculated for all relevant hazards, MU Level 2 and MU Level 3 results were developed only for illustrative scenarios. For these reasons, the Level 3 PRA project's ISR task focused on estimating MU and multi-source risk only. In other words, no specific calculations for ISR were performed.[5]

1.3 Terminology for ISR

Because MUPRA and ISR (and the associated multi-source risk) are new tasks within PRA, there is limited consensus on acceptable terminology and acronyms for these tasks. To the extent possible, the terminology used in this report is consistent with other references (e.g., IAEA, 2019; EPRI, 2021a). In some cases, this report has defined new terms or acronyms not identified elsewhere. Section 1.2 identified and defined some of these terms.

Some sites may have reactors of different designs (e.g., different vendors), creating the possibility of multiple MUPRAs being created for a single site (e.g., one MUPRA for two BWRs

[5] In principle, ISR results could be developed for the illustrative scenarios at the individual initiator level only (e.g., for the seismic bin 6 initiating event) because all the necessary inputs are available. However, as noted above, such results would not provide any insights that the single source PRAs do not already provide.

on site and another MUPRA for two PWRs on site). The reference site for the L3PRA project is the simplest of situations (i.e., two nearly identical reactors; two connected SFPs; one DCS facility). However, it should be noted that expressing PRA results for a "site" could be considered ambiguous because it does not specify the number of radiological sources (e.g., NPPs) on the site. Therefore, sitewide risk results should be carefully defined for each site modeled.

In principle, two categories of terms are needed for MUPRA, ISR, and multi-source risk: (1) those related to sitewide risk results, and (2) those related to MU risk results. Since the ISR task for the L3PRA project focused mainly on MUPRA, no new terms for ISR or multi-source risk were defined. The following is a list of MUPRA terms used in this report. This terminology is generally consistent with the IAEA reports (IAEA, 2019; IAEA, 2021a) and EPRI report (EPRI, 2021a) on MUPRA.

SUPRA	(traditional) single-unit PRA
SUCDF:	single-unit core damage frequency per reactor-critical-year (as typically calculated by Level 1 PRA for a single unit)
IEF	Initiating event frequency per reactor-critical-year (as typically calculated by Level 1 PRA for a single unit); also "SUIEF"
SOCDF	Frequency <u>per reactor-critical-year</u> of an accident involving core damage to ONLY single units on a multi-unit site
MUPRA	multi-unit PRA for two or more (typically similar or identical) reactors on a site
MUIE	multi-unit initiating event (also, could be called a "sitewide IE")
MUIEF	multi-unit initiating event frequency <u>per reactor-critical-year</u> for two or more reactors on the site
MUCDF	multi-unit core damage frequency <u>per reactor-critical-year</u> of an accident involving core damage to two or more reactor units on a multi-unit site
MURCF	multi-unit release category frequency per reactor critical-year of an accident involving two releases from two or more reactor units on a multi-unit site

A traditionally performed single-unit PRA (SUPRA) addresses, for example, the full-scope initiating events (IEs) identified as relevant to the plant site. However, some of these IEs would cause both units on a two-unit plant site to experience a reactor trip. These IEs are called multi-unit IEs (MUIEs). Examples of such MUIEs include seismic events and certain losses of offsite power (LOOPs). Consequently, SUPRAs already include accident sequences that could lead to simultaneous core damage for both reactors. Initiating events that do not cause a concurrent reactor trip for both units are not MUIEs. Examples of such IEs are certain internal events such as steam generator tube ruptures (SGTRs).

MUCDF is calculated only for MUIEs. However, as for SUPRAs, MUPRAs must account for additional failures beyond the MUIE in order to calculate MUCDFs. While MUCDF can occur via sequences involving an MUIE followed by separate random failures in each unit that lead to core damage at both units, the probabilities of such simultaneous random failures involving both units are generally very low. Consequently, similar to SUPRA modeling, dependencies or "coupling" between the two units are also represented in MUPRAs and the dominant MU cutsets typically contain such dependent failures. As a simple example, one of the dominant cutsets for all LOOP events in the SUPRA in the L3PRA project contains only two elements: (1) the LOOP IEF, and (2) the common cause failure (CCF) of both (i.e., two-of-two) emergency diesel generator (EDG) load sequencers. The parallel MU cutset also would contain two elements: (1) the LOOP MUIEF, and (2) the multi-unit CCF of all EDG load sequencers (i.e., four-of-four).

As discussed in Section 1.2, Figure 1-1 provides a Venn diagram that illustrates the relationship between the results for a traditional single-unit PRA and a multi-unit PRA for two, identical reactor units. In Figure 1-1, the area within a complete circle represents the total single-unit CDF that is calculated by traditional PRAs (i.e., SUCDF). More importantly for integrated site risk, this diagram shows the area of overlap that represents both units experiencing core damage simultaneously (i.e., MUCDF). Therefore, SOCDF is represented by the area outside of the overlapping circles (i.e., accident sequences that involve only one unit experiencing core damage). This Venn diagram is more than a visual aid. If there are no new accident sequences identified when both units are considered together,[6] then this Venn diagram also represents the Boolean algebra that is needed to calculate MUCDF.

As noted in Section 1.2 "overlap" between the two reactors and the SFP (i.e., accidents involving both reactors and the SFP). As for Figures ES-1 and ES-2, the estimation of ISR for the two reactors and the SFP would be overestimated if the single source risks were simply added (i.e., dependencies between the radiological sources were ignored).

Appendix A provides risk equations for two identical units (which is consistent with the reference site for the L3PRA project), representing relationships between the two units consistent with that shown in Figure 1-1.

Finally, it can be informative to compare MU risk results to those from the SUPRA. For example, the IAEA reports (IAEA, 2019; IAEA, 2021a) define terms to make such comparisons (e.g., the conditional probability of an accident involving multiple units on the site given core damage at a single reactor unit).

1.4 Development of Multi-Unit and Multi-Source Results

The ISR task was performed principally using the results of the L3PRA project's PRAs and underlying plant-specific information that are documented in other project reports. The expertise and experience of the L3PRA project team also were important to the development of the ISR approach and its implementation. Also, as noted above, the Level 3 PRA project's ISR task was focused on calculating multi-unit and multi-source risk (as opposed to complete, or total, site risk).

[6] The sitewide dependency assessment performed for the L3PRA project addressed this potential concern. See Section 4 and Appendix G for more details.

Some key aspects of the ISR task performance are:

- When new acronyms or terms were defined, they were chosen to be consistent with and/or extensions of existing acronyms and terms.
- Both base case (i.e., the PRA models based on the 2012 reference plant) and 2020-FLEX sensitivity case results for the reactors were used when appropriate.
- Because a low power and shutdown reactor PRA was performed for only internal events, and seismic events were anticipated to be a significant contributor to multi-unit and integrated site risk, the scope of the ISR task only includes when both reactors are at power.
- The sitewide dependency assessment approach used by the L3PRA project is an expansion of the approaches described in the IAEA reports (IAEA, 2019; IAEA, 2021a) and EPRI report (EPRI, 2021a).
- Consistent with existing guidance for MUPRAs (see, for example, EPRI [2021a]), a simplified approach to developing MU risk results was used because only limited dependencies between the two reactors on the reference site were identified. This simplified approach did not require formal, fault tree-event tree modeling. Instead, for example, Excel spreadsheets that captured the Level 1 PRA cutsets were used to estimate multi-unit core damage frequencies (MUCDFs).
- In order to represent dependencies between the reactors, various generic coupling factors were used. These factors were developed to be consistent with the treatment of dependencies in the single-unit PRA (SUPRA) and with the current state-of-practice for multi-unit dependencies. The generic factors selected are expected to be conservative.
- MUCDF estimates for all hazards were developed using single-unit cutsets from the L3PRA project's at-power PRAs. Identified MU dependencies were represented in these calculations.
- In order to conserve resources and address computing limitations, the existing L3PRA project's models were used to develop multi-unit (MU) Level 2 PRA and MU Level 3 PRA results for only two illustrative multi-source scenarios.
- Similarly, Level 3 PRA consequence results were developed for only one illustrative scenario that involved all relevant radiological sources on the reference site. In practice, this scenario involved both reactors and the hydraulically connected SFPs (i.e., a multi-source scenario).

1.5 Report Organization

This report is organized in 12 sections:

- Section 1 provides an overview and background on the ISR task for the L3PRA project.
- Section 2 describes the technical approach for the ISR task.

- Section 3 describes the scope, screening criteria, and risk metrics used by the ISR task.
- Section 4 describes the sitewide dependency assessment used by the ISR task.
- Section 5 describes the multi-unit and sitewide initiating events analysis.
- Section 6 describes the multi-unit Level 1 PRA risk contributions needed to support the ISR task.
- Section 7 describes an illustrative, multi-source risk scenario that involves both reactors and other relevant radiological sources (e.g., the SFPs) that require concurrent accident mitigation.
- Section 8 describes the multi-unit Level 2 PRA contributions needed to support the ISR task for two illustrative scenarios.
- Section 9 describes: (1) the multi-unit Level 3 PRA contributions needed to support the ISR task for two illustrative scenarios, and (2) the combined multi-unit and other radiological sources contributions to Level 3 PRA risk for one illustrative multi-source scenario.
- Section 10 describes the key sources of uncertainty for the ISR task.
- Section 11 provides an overall summary and conclusions for the ISR task.
- Section 12 provides the references used in this report.

In addition, this report includes 11 appendices:

- Appendix A provides key risk equations that are needed for multi-unit PRA and the ISR task.
- Appendix B describes the approach for sitewide dependency assessment used by the ISR task.
- Appendix C describes the identification and analysis of multi-unit and sitewide initiating events needed to support the ISR task.
- Appendix D describes the Phase 2 assessment of sitewide dependencies needed to support the ISR task.
- Appendix E describes the identification of potential cross-source common cause failures needed to support the ISR task.
- Appendix F describes the human and organizational sitewide dependencies identified for the ISR task.
- Appendix G describes the identification of other Phase 3 sitewide dependencies needed to support the ISR task.

- Appendix H provides background on the coupling factors used for calculating contributions to multi-unit Level 1 risk.

- Appendix I provides a more detailed description of the multi-unit Level 1 risk estimations used to support the ISR task.

- Appendix J describes the testing of the cutset estimation method for multi-unit Level 1 results needed to support the ISR task.

- Appendix K provides a more detailed description of the multi-unit Level 2 contributions needed to support the ISR task.

Simplified diagrams for key systems are provided in Volume 2 of this NUREG series (NRC, 2022a).

2 TECHNICAL APPROACH FOR INTEGRATED SITE RISK

Although the NRC's Level 3 PRA study is generally being performed consistent with current standards and state-of-practice using existing PRA technology, there are some technical elements that necessitate methodological development due to a lack of sufficient experience to define a current state-of-practice. One such technical element is the Integrated Site Risk (ISR) task.

The original objectives of the ISR task were to:

- Estimate the integrated site risk for the reference site.
- Identify and characterize significant contributors to ISR for the reference site.

However, as further discussed in Section 3.1, the objectives of the ISR task for the L3PRA project were modified to the following:

- Estimate the multi-source risk for the reference site.
- Identify and characterize significant contributors to multi-source risk for the reference site.

2.1 <u>Motivation and Background for ISR Technical Approach</u>

The technical approach for performing ISR has evolved over the course of the L3PRA project (see, for example, Hudson, 2018 and 2019). PRA results for different hazards and operating modes, newly published reports on MUPRA, changing computational capabilities, and continuing technical discussions among the project team all contributed to the development of the final ISR approach. Performance and results from some of the ISR tasks, such as a sitewide dependency assessment and trial MU risk estimations, also were important influences on the final implementation of the ISR approach.

Particularly key factors in the development of the ISR approach are:

- NRC's SAPHIRE PRA software is more limited in its ability to handle large PRA models than some industry software (e.g., CAFTA). Consequently, using formal logic models to make MU risk calculations was not possible. However, some small pilot models were run to explore differences in results from linked PRA models versus the simpler approach used in the L3PRA project.

- The IAEA (IAEA, 2019; IAEA, 2021a) and EPRI (EPRI, 2021a) reports were completed and available to the L3PRA project team in time to inform the development of the ISR approach and its implementation. In particular, information on sitewide dependencies and the state-of-the-art for various technical aspects of MUPRA (e.g., modeling inter-unit common cause failures and human and organizational dependencies) were important inputs to the L3PRA project's ISR task.

- Most of the Level 1 and Level 2 PRAs for the Level 3 PRA project were completed when final decisions were made on the ISR approach. The results of these PRAs were valuable inputs to making risk-informed decisions on the ISR approach. For example, the project team noted that, for almost all hazards, Level 1 PRA cutsets representing

95 percent of CDF could be extrapolated to give a reasonable approximation of the total CDF, thereby greatly reducing the number of cutsets that need to be analyzed.

- Results of sitewide dependency assessment in support of the ISR task (see Section 3 for the steps in the ISR task and Appendix C through Appendix G for detailed results of the sitewide dependency assessment) demonstrated that there are very few dependencies between the two reactors on the reference site and that SFPs are generally independent of the reactors. Because there are so few dependencies to represent in the ISR task, a simplified approach to determining MU risk, and ISR overall, was determined to be justified.

- As noted in Section 1.1 (see, especially, Bixler and Kim [2021]), computational limitations for performing multi-unit Level 2 PRA are well known. Consequently, the L3PRA project team made simplifications in its approach to developing MU Level 2 PRA results. This also resulted in limitations in the number of multi-source scenarios developed for MU Level 2 PRA and Level 3 PRA, as well as the overall ISR Level 3 PRA consequence results.

- The EPRI report (EPRI, 2021a) identified some limitations in the state-of-the-art for MUPRA. For example, there is a lack of sufficient data to support the development of some MU coupling factors or other MU dependency representations. Consequently, the L3PRA project team chose to use generic coupling factors for inter-unit common cause failures and made other simplifying modeling choices.

- Like MU Level 2 PRA, there is very limited experience in performing MU Level 3 PRA.

- The completion schedule for the overall L3PRA project influenced several decisions on the development of the ISR approach and its implementation. In particular, certain technical tasks were delayed or not performed (e.g., performance of the "FLEX sensitivity" case for the SFPs), such that these results were not available for use in the ISR task. Also, the scope of the ISR task was revised to support completion of the overall L3PRA project.

2.2 Overview of Technical Approach

The steps to perform the ISR task are:

- Step 1: Specify scope, screening criteria, and risk metrics
- Step 2: Perform sitewide dependency assessment
- Step 3: Perform MU (and sitewide) initiating events analysis
- Step 4: Estimate MU Level 1 CDF
- Step 5: Identify and define an illustrative multi-source scenario
- Step 6: Develop and quantify MU Level 2 PRA
- Step 7: Develop and quantify MU Level 3 PRA
- Step 8: Combine MU risk with risk from SFPs and DCS
- Step 9: Identify key sources of uncertainty

Note that, consistent with discussions provided above, the Level 3 PRA project's ISR task is focused on calculating MU and multi-source risk; ISR, per the definitions provided above, is not calculated. Calculation of ISR would require additional complex steps.

For the Level 3 PRA project, Step 4 was first performed for the at-power, internal events Level 1 PRA, then repeated for each hazard group (both internal and external). As described in the sections below, MUCDF results were generated by using single unit PRA cutsets and a cutset estimation method that was developed by the L3PRA project ISR team.

In recognition of limited project resources and limited computational ability for Level 2 PRA, Step 5 was added to identify and define a scenario in which both reactors and other relevant radiological sources (e.g., the SFPs) required accident mitigation. The results of Step 5 were used to focus Steps 6 and 7 (i.e., MU Level 2 and Level 3 PRA) on only two MU scenarios. Also, the results of Step 5 were used to focus Step 8 on a single illustrative multi-source scenario for which MU risks were integrated with those from the other major onsite radiological sources (i.e., the SFPs and DCS).

The subsequent sections in this report (along with associated appendices) provide further details on the steps of the ISR task.

3 SCOPE, SCREENING CRITERIA, AND RISK METRICS FOR INTEGRATED SITE RISK

This section summarizes the first step in the ISR task. It should be noted that this step was performed iteratively as the ISR task progressed.

3.1 Scope of ISR Task

The first step of the ISR task is to specify the scope of the risk calculations and associated PRA models that are to be developed for the selected NPP site. This requires choosing from available options for each of four main PRA scope elements: (1) radiological sources and operating configurations; (2) IE hazard groups; and (3) PRA end-states and risk metrics, and (4) ISR calculations.

In general, the ISR task is subject to the same scope, limitations, and assumptions as the larger L3PRA project. For example, it should be noted that the overall freeze date for reference plant information used in the L3PRA project is August 2012 (with a few exceptions). However, sensitivity analyses for FLEX strategies (which were implemented by the U.S. nuclear power industry after 2012) have been performed for the two reactors.

The scope of the overall 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. Fresh nuclear fuel, radiological waste, and minor radiological sources (e.g., calibration devices) are not included as part of the scope. In addition, deliberate malevolent acts (e.g., terrorism and sabotage) are excluded from the scope of this study. Other important project-wide limitations in scope include:

- The L3PRA project only addresses reactor low power and shutdown risk for internal events.
- The L3PRA project did not address seismically-induced fires quantitatively.
- The following events were assessed as being negligible risk contributions and, therefore, were screened out of further consideration by the L3PRA project:
    - cross-unit internal floods for the control buildings
    - various external hazards, including external floods

Using the overall L3PRA project scope, Table 3-1 summarizes options available for the ISR task for the first three L3PRA scope elements. The ISR approach used for the L3PRA project could be used to address all the available options (including the different plant operating states) shown in Table 3-1. However, only the shaded options shown in Table 3-1 were addressed by the ISR task. Those options that were not addressed (e.g., low power and shutdown for the two reactors) were omitted due to project resource limitations.

Table 3-1 Multi-Source PRA Model Scoping Options for Integrated Site Risk

PRA Scope Element	Scoping Options	
Radiological Sources	Operating Reactor Units	
	Operating SFPs	
	Dry Cask Storage Facility	
Radiological Source Operating Configurations	Operating Reactor Units	At-Power
		Low Power and Shutdown
	Operating SFPs	Nominal
		Refueling Outage States
		Cask Loading
	DCS	Storage
		Cask Loading
IE Hazard Groups	Internal Hazards	Internal Events
		Internal Floods
		Internal Fires
	External Hazards	Seismic Events
		High Winds
		Other External Hazards
PRA End-States and Risk Metrics	Level 1: Nuclear Fuel Damage	Fuel Damage Frequencies
	Level 2: Radiological Release	Radiological Release Frequencies
	Level 3: Offsite Radiological Consequences	Frequencies of Offsite Radiological Consequences

Some scope issues are important only for the ISR task. Scope limitations specific to the ISR task include:

- The ISR task addressed only at-power conditions for both the reactors and the SFPs.
- The ISR task did not address combinations of only one reactor unit with either the SFPs or dry cask storage.
- The ISR task addressed only one of the two cases that were performed specifically for the SFP PRAs—the SFP base case.[7] As documented in the SFP PRA report (NRC, 2025a), two analyses were performed for the SFPs: (1) the base case for scenarios that lead to SFP uncovery within 7 days, and (2) a sensitivity case that relaxes the 7-day truncation. For the base case, the only events that lead to uncovery within 7 days are

[7] Note that the “SFP base case” is different than the L3PRA project’s “base case” that relates to the freeze date of August 2012.

those that result in inventory loss through a leak or sloshing out of the SFPs (i.e., mostly seismic events and also a reactor-side loss of inventory loss (LOINV) with the gates open).

- Some types of dependencies between the reactors or between the reactors and the SFPs or DCS can be difficult to assess (e.g., due to limitations in the availability of information or the state-of-the-art for PRA or hazard modeling). Therefore, analysts identified such dependencies in a way similar to that for the identification of sources of uncertainty.

- Formal sensitivity analyses related to FLEX strategies for the SFP study were not available for the ISR task.

- MU results for MU Level 1 PRA addressed the full set of identified MUIEs. However, consistent with limitations in the current state-of-the-art for MU Level 2 and Level 3 PRA (including generic PRA software limitations), the ISR task developed MU Level 2 and Level 3 results for selected release categories and risk metrics for only two MUIEs (and sitewide IEs):

    - weather-related LOOPs
    - seismic bin 6 events

- Level 3 consequence results were developed for a single, illustrative multi-source scenario, seismic bin 6, combining multi-unit risk with SFP risk.

Some of the scope limitations listed above also limited the Level 3 PRA project's options for ISR calculations. In particular, the second bullet above states that ISR task did not address combinations of only one reactor unit with either the SFPs or dry cask storage. Consequently, per the definition of ISR given in Section 1 and the Venn diagram shown in Figure 1-2, the risk associated with the SFPs and a single reactor cannot be calculated. Therefore, ISR could not be calculated as defined in Section 1.2. Instead, the ISR task focused on multi-source risk resulting from simultaneous accidents for both reactors and the SFPs. As noted in Section 1.2, MU risk and multi-source risk represent the most challenging scenarios for accident response. As such, MU and multi-source risk results can provide additional insights and identify potential vulnerabilities that the single source PRAs and ISR results cannot provide.

At present, these scope limitations represent potential future research topics, including additional potential advances in the state-of-the-art for MUPRA and integrated site risk.

Screening was also used to further streamline the ISR task. This screening analysis is discussed in the next section.

3.2 Specify Screening Criteria

Although the scope selected for the ISR task in the L3PRA includes all the shaded items depicted in Table 3-1, screening has been used to focus on the relevant contributors to MU and multi-source risk and to simplify (e.g., limit the size and complexity of) the risk models, reducing the staff and computing resources needed. Screening is a common practice in the performance of traditional PRAs. Risk-significance is the most common criteria for such screening using risk importance measures, such as the Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures. Screening is also recognized as an important part of developing multi-unit risk models in other approaches (e.g., IAEA, 2019; 2021a; EPRI, 2021a). Consequently, screening criteria used in other approaches, both qualitative and quantitative, have been adopted for the L3PRA project's ISR effort. However, additional screening choices were made during the development of MU and multi-source calculation approaches in order to control the size, complexity, and duration of the ISR task.

For the L3PRA project, the ISR task uses both qualitative and quantitative screening in several steps of the process. Examples of screening used in the ISR task include:

- Multiple screening approaches were used in the assessment of site dependencies, such as:
    - Screening out a hazard group or IE from the set of potential multi-unit IEs if all three of the following qualitative criteria (from page 19 of IAEA [2021a]) are met:
        1. The event does not immediately result in a trip of multiple units.
        2. The event does not result in an immediate trip of one unit and a degraded condition at another unit that will eventually lead to a trip.
        3. The event does not result in a degraded condition at multiple units that will eventually lead to a trip of the units.
    - Carrying over screening assumptions used for other L3PRA project PRA models. For example, when assessing potential dependencies related to shared systems, structures, and components (SSCs), using Level 1 internal flooding PRA assumptions regarding the potential for internal flooding in shared structures.
    - Using single-unit (SU) PRA importance measures and SU cutset reviews to select which common cause failures (CCFs) to model as inter-unit (or multi-unit) CCFs.
- When developing MUCDF estimates, reviews of the SU Level 1 internal events cutsets and the assessment of potential human and associated organizational dependencies for the reference site indicated that multi-unit CCFs are likely more important than potential multi-unit dependencies affecting human failure events (HFEs). Consequently, efforts to represent cross-unit or MU CCFs were emphasized over representation of potential human and organizational dependencies for the reference site.
- In order to select and develop multi-source accident sequences, the ISR task focused on the sitewide initiating events and scenarios involving dependencies between the reactors and SFPs, since these were expected to be most significant to multi-source risk. For example, the multi-source Level 2 and 3 PRA risk calculations focused on seismic events, since these events can have a significant impact on both the reactors and the SFPs.

3.3 Risk Metrics

The definition of risk metrics is important to any PRA project in order to effectively communicate its results. Risk metrics for the overall L3PRA project have already been provided in other reports. This report focuses on risk metrics needed specifically for the ISR task.

The IAEA reports on MUPRA (IAEA, 2019; IAEA, 2021a) and an IAEA technical report on risk aggregation (IAEA, 2021b) provide discussions on appropriate risk metrics for multi-unit and sitewide risk results. The selection of risk metrics for the L3PRA project's ISR task is based on these reports as well as the specific requirements and constraints of the L3PRA project.

Section 1.3 already defined the key risk metrics for multi-unit Level 1 and 2 PRA results developed in the L3PRA project's ISR task:

- for multi-unit Level 1 PRA: multi-unit core damage frequency (MUCDF)
- for multi-unit Level 2 PRA: multi-unit release category frequencies (MURCFs)

No new Level 3 PRA metrics were selected, as the existing set of L3PRA project Level 3 PRA risk metrics inherently capture the site risk. A subset of the offsite public risk metrics used in other L3PRA reports are reported for the ISR task. Specifically, four consequence measures were selected that are typically used in consequence analyses supporting regulatory analyses and that are considered adequate to demonstrate potential methodologies for multi-unit risk assessment:

- population-weighted risk to an individual within 1 mile of the site boundary of an early fatality (early fatality risk/year)
- population-weighted risk to an individual within 10 miles of the site of a latent cancer fatality (cancer fatality risk/year)
- population dose risk integrated across the population within 50 and 100 miles of the site (person-rem/yr)
- economic cost risk integrated across the region within 50 and 100 miles of the site ($/year)

The above risk metrics were chosen to:

- be consistent with current regulatory applications (i.e., individual risk of early and latent fatalities, collective dose, and economic impacts)
- allow comparison with previous studies (i.e., total early and latent fatalities)

The ISR task performed multi-source calculations for Level 2 and Level 3 PRA. While the Level 1 PRA risk metrics of core damage frequency for reactors and significant fuel uncovery frequency for the SFPs may be close in meaning, the ISR task did not produce Level 1 PRA multi-source results.

4 SITEWIDE DEPENDENCY ASSESSMENT

This section describes the sitewide dependency assessment that was performed as the second step in the overall ISR task. The L3PRA project team made sitewide dependency assessment a formal, separate task. The L3PRA project's sitewide dependency assessment task expands upon a similar task called the "initial site assessment" task in EPRI (2021a). The sitewide dependency assessment performed for the ISR task expanded on the more familiar dependency analyses performed for traditional PRA. For example, the sitewide dependency assessment:

- addressed all radiological sources on the reference site (e.g., cross-unit or inter-unit dependencies rather than just intra-unit dependencies)

- addressed types of dependencies that may not typically be addressed in traditional PRA (e.g., dependencies between radiological sources regarding the operator actions and organizations responsible for accident mitigation)

- identified potential sitewide dependencies and assessed whether, and how, to represent such dependencies in MU and multi-source risk calculations

- Note that the last bullet above implies that not all identified potential sitewide dependencies were represented in MU and multi-source risk calculations. In some cases, relative risk, risk importance measures, or analyst judgment supported the omission of a potential sitewide dependency from MU and multi-source risk calculations.

4.1 Introduction

There are always dependencies between radiological sources (e.g., dependencies between the two reactor units) on a nuclear power plant (NPP) site. The EPRI report on multi-unit (MU) risk (EPRI, 2021a) discusses how even "independent" reactor units on a shared site have some unavoidable dependencies (e.g., shared initiating events and shared power sources). Such sitewide dependencies between radiological sources need to be addressed and, for this reason, the individual contributions from the separate PRAs performed for the L3PRA project are not sufficient alone to perform ISR tasks. Also, if sitewide dependencies are not identified and addressed appropriately then MU and multi-source risk will likely be underestimated.

For the Level 3 PRA (L3PRA) project, MU and multi-source risk was estimated by aggregating, while accounting for sitewide dependencies, the risk contributions from modeled accident scenarios for the major radiological sources on the selected reference NPP site, that is:

- two operating reactor units (Unit 1 and Unit 2)
- two spent fuel pools (SFPs), one for each operating reactor unit
- a dry cask storage (DCS) facility

For example, accident scenarios that could involve core damage for both reactor units were considered, as well as simultaneous failures of reactor unit(s) with other radiological sources (e.g., spent fuel pool(s)).

Section 4.2 describes the categories of sitewide dependencies considered in the ISR task. These categories were used to help identify potential sitewide dependencies. In addition, different categories of sitewide dependencies may be accounted for differently in calculating multi-source risk.

The approach for performing the sitewide dependency assessment is summarized in Section 4.3. Appendix B provides further details on how the sitewide dependency assessment was performed to support the ISR task. Appendix B also serves as a stand-alone guidance document for performing a sitewide dependency assessment. As such, Appendix B is preserved in its entirety despite containing repetition of the information provided in this section.

The results of the sitewide dependency assessment are summarized for the Phase 1, 2, and 3 sitewide dependencies in Section 4.4, Section 4.5, and Section 4.6, respectively. Detailed results of the Phase 1, 2, and 3 sitewide dependency assessments (e.g., identification of multi-unit and sitewide initiating events) are provided in Appendix C through Appendix G.

The discussions of MU and multi-source calculations (see Sections 5, 6, 7, and 8 and Appendix H, Appendix I, and Appendix K) address how the results of the sitewide dependency assessment were used.

4.2 Categories of Sitewide Dependencies

The L3PRA project's ISR approach uses a categorization scheme to identify, characterize, and document the sitewide dependencies for the selected NPP site. This categorization scheme supports the systematic search for dependencies by recognizing the different ways dependencies can impact structures, systems, and components (SSCs) and operator actions. In addition, different categories of dependencies may be accounted for differently in calculating MU and multi-source risk. The specific categorization scheme that was used is a combination of similar schemes used in IAEA (2019, 2021a) and EPRI (2021a) guidance for MUPRA.

Table 4-1 provides high-level definitions of each dependency category, some illustrative examples, and expected ways that such dependencies would be represented in risk calculations. The definitions and illustrative examples were selected to guide analysts in assigning an identified dependency to a category because some types of dependencies might be interpreted to belong to multiple categories (i.e., some overlaps between categories may exist). Later sections provide brief descriptions of each category and guidance on their identification and representation.

The different categories of potential sitewide dependencies also were used to divide the assessment of sitewide dependencies into three phases, as discussed in the next section.

4.3 Approach for Sitewide Dependency Assessment

The assessment of sitewide dependencies was performed in a phased approach for the different categories of potential dependencies shown in Table 4-1. Three phases have been defined as follows:

- Phase 1 Assessment:
    - sitewide and multi-unit initiating events

- Phase 2 Assessment:
    - shared physical resources
    - shared or connected systems, structures, & components (SSCs)

- Phase 3 Assessment:
    - identical components (e.g., expansion of CCF groups)
    - proximity dependencies
    - human or organizational dependencies
    - accident propagation between units
    - potential hazards correlations

A phased approach offers three benefits:

1. It focuses on the more obvious and more easily represented sitewide dependencies first.

2. It allows the use of earlier potential sitewide dependency results to inform the rigor of later sitewide dependency assessments.

3. It provides a general understanding of the “coupling” between reactors (and between the reactors and the SFPs, and between the reactors and the DCS facility) that determines the complexity in calculating MU risk and multi-source risk in later ISR tasks.

Table 4-1 Potential Multi-Unit and Other Sitewide Dependencies

Category	Definition	Example(s)	How Modeled
Sitewide and Multi-Unit IEs	IEs that impact multiple reactor units and/or multiple radiological sources on site.	Losses of offsite power that are grid-, switchyard-, or weather-related.	Risk models are constructed to represent such IEs as causing reactor trip in both units concurrently.
Shared Physical Resources	Resources available to provide common support to reactor units, spent fuel pools, and dry cask storage facility.	Electric power via common grid and/or switchyard, ultimate heat sink, intake structure, water supplies for fire protection, diesel fuel, etc.	Common electrical grid and switchyard could be identified in this category but should be addressed under the category "sitewide and multi-unit IEs." Other shared resources that could be identified (e.g., common water, diesel fuel) can be addressed in risk models.[8]
Shared or Connected SSCs	Shared or cross-tied systems and components that support multiple radiological sources under various conditions.	The service water system or a "swing" diesel generator may be shared by both reactor units and other radiological sources.	Like the "shared physical resources" of water and fuel, shared systems or components can be addressed in risk models (using flags or similar tools) after developing a scheme for prioritizing which radiological source is supplied first (or only).[9]
		For some NPPs, there may be an alternate alignment (e.g., cross-tie) of a system or component such that it can support the alternate unit (e.g., Unit 2 emergency diesel generator (EDG) can be cross-tied to feed Unit 1).[9]	Logic models and basic event naming for systems and components can be altered to account for cross-tied equipment.
	Shared or connected structures that support multiple radiological sources.	Two reactor units may share structures (e.g., turbine building) or may be connected (e.g., main control rooms for both units are connected).	Shared main control room (MCR) is a special case that should be addressed by HRA. Other shared or connected structures should be treated in a manner consistent with the hazard group (e.g., internal or external floods, internal fires, seismic event) that addresses the structures.

[8] Loss of these resources can be accommodated in the logic models for all impacted radiological sources by using the same basic event names in all the models. If these resources remain available, a priority scheme can be used for common supplies that designates reactor unit 1, for example, as getting all supplies it needs first, reactor Unit 2 as having secondary priority, and so on, until known supplies are depleted. Flag sets or similar PRA modeling techniques might be used to select which radiological source is credited with adequate physical resources (versus those that are not given such credit). Different time frames that are associated with different strategies (e.g., implementation of FLEX or Severe Accident Management Guidelines (SAMGs)) may imply different requirements and availability of physical resources.

[9] The Level 2 HRA effort explored crediting the EDG cross-tie for the reference NPP but learned that: (a) it is not formally proceduralized, and (b) while operators and decision-makers are aware of this potential capability, it is not likely to be used since two reactor units could end up without alternating current (AC) power if the cross-tie is done improperly. Also, FLEX strategies and associated equipment are available now, making use of this option even more unlikely.

Table 4-1 Potential Multi-Unit and Other Sitewide Dependencies (cont.)

Category	Definition	Example(s)	How Modeled
Identical Components	Components that have the same design, maintenance, operation, and operating environment for multiple units.	Failure of similar components installed in each unit due to common-cause.	Such dependencies can be addressed in risk models as potential inter-unit common-cause failures (CCFs).
Proximity Dependencies	Dependencies that arise across radiological sources from: (1) exposure of multiple SSCs to shared phenomenological or environmental conditions, (2) common features between units, or (3) operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source.	Failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat or cold or radiation levels), debris, explosions, etc., from a nearby radiological source. External hazard fails identical or similar structures due to common location of structures for both units.	These dependencies are not likely to have been identified in individual risk models for each radiological source. External hazards and radiological concerns (e.g., Level 2 PRA) are likely to be the principal concern for SSCs in both units that would share phenomenological or environmental conditions. Environmental conditions that impact operator actions can be modeled similarly to SSCs but should be addressed as “human or organizational dependencies.”[10] External hazards are likely to be of most concern for common features between units.
Human or Organizational Dependencies	Dependencies between operator actions across multiple radiological sources that can result from multiple causes, including sharing of staff and shared organizational factors.	Common training, procedures, human machine interface, or command and control structure cause recovery actions taken in response to an accident affecting one radiological source to be dependent upon those taken in response to an accident affecting another radiological source. Also, some staff (e.g., field operators, fire brigade, health physics, technicians) may be shared by all radiological sources on the site.	If dependencies related to common training, procedures, human machine interface, etc., need to be addressed, the HRA for each unit should be adjusted. Impacts from limited staffing, both in the Technical Support Center (TSC) and for field operators, can be represented by adjusting HEPs in the multi-unit model with Unit 1 getting full credit and Unit 2 receiving reduced or no credit. It is not expected that the shared offsite technical support will be modeled.
Potential Accident Propagation Between Units	A reactor trip or subsequent failures on one reactor might cause an event for another radiological source on site.	Propagation of an accident from one radiological source to another is more likely if, for example, two units share systems or components, or are connected in some other way. Also, if conditions cause an automatic trip in one unit, that may lead to a manual trip in another.	Such dependencies may not have been identified in original PRA models.[11] Once they are identified, such dependencies can be addressed in logic models with, for example, Unit 2 failure being conditional on a certain Unit 1 failure(s).

[10] The timing of the conditions from one reactor that can affect another reactor is also important. Once such dependencies are identified as impacting SSCs or operator actions, one approach would be to assign conditional failure probabilities to basic events or HFEs (e.g., the basic events and HFEs in Unit 2’s risk model can be altered due to failures, environmental conditions, debris, explosions, etc., that exist for nearby Unit 1).

[11] The EPRI guidance (2021a) suggests that such “cascading” dependencies only occur if there are cross-connected systems. Therefore, this category may not be important to the reference site as there are few such dependencies.

Table 4-1 Potential Multi-Unit and Other Sitewide Dependencies (cont.)

Category	Definition	Example(s)	How Modeled
Potential Hazards Correlations	SSCs and operator actions may be affected in the same or similar ways by external hazards (e.g., seismic or external floods).	Simultaneous (or nearly simultaneous) failures of SSCs or operator actions for both reactor units may occur due to impact of a seismic event.	Such dependencies are likely to be already addressed in the PRA for each radiological source and for the relevant hazard (e.g., the same seismic hazard curve and seismic correlations are used for both reactor units).

A summary of the phase-specific sitewide dependency assessment guidance is given below. More detailed guidance is given in Appendix B. Appendix B is intended to be a stand-alone source of guidance for performing a sitewide dependency assessment.

For each phase in the sitewide dependency assessment, the assessment was performed in this order:

- potential dependencies for the two reactors were identified first
- potential dependencies between the SFPs and the reactors were identified second
- any dependencies between the DCS and other radiological sources (i.e., the reactors and SFPs) were identified last

Due to resource limitations, this approach did not address:

- combinations of only one reactor unit with either the SFPs or dry cask storage
- any combinations that do not involve both reactors
- any plant operating states beyond at-power operations

4.3.1 Phase 1 Guidance for Sitewide Dependency Assessment

Phase 1 assessment addresses sitewide initiating events and MU initiating events (MUIEs) (shown as the first category of dependencies in Table 4-1). Initiating events (IEs) that impact multiple reactor units and/or multiple radiological sources on site are expected to be especially important to investigating MU and multi-source risk. In particular, the early identification of each IE can help to focus attention on only the relevant portions of the individual risk models that need to be incorporated into MU and multi-source risk calculations. Sometimes this kind of IE is called a “common cause initiator (CCI).”

Sitewide initiating events are identified with the following steps:

1. identify Level 1 PRA MUIEs for the two reactor units
2. identify MUIEs that also impact the SFPs
3. identify MUIEs that also impact dry cask storage

Note that this approach first identified IEs that are important to the reactors, then assessed whether these same IEs are important to the SFPs and dry cask storage. In other words, the focus of the ISR task is on scenarios involving both reactors and not on scenarios involving only one reactor and either the SFPs or DCS.

In all cases, a Level 1 PRA IE can be screened out from consideration as an MUIE if all three of these screening criteria are true:

1. The event does not immediately result in a trip of both units.
2. The event does not result in an immediate trip of one unit and a degraded condition at another unit that will eventually lead to a trip (including a required manual trip).
3. The event does not result in a degraded condition at both units that will eventually lead to a trip of the units (including required manual trip(s)).

Note that the application of these criteria requires some understanding of the timing of when a reactor trip occurs. For this step in the ISR task, the working assumption is that the three screening criteria would be met unless both units experience a plant trip nearly simultaneously. Later ISR steps examine the timing of multi-source events more closely.

When considering the SFPs and DCS in the L3PRA project, the L3PRA single source PRA results were used to identify which MUIEs could lead to risk-significant consequences that would contribute to MU and multi-source risk.

Section 12B.5 provides additional guidance for the identification of Phase 1 sitewide dependencies.

4.3.2 Phase 2 Guidance for Sitewide Dependency Assessment

The assessment of potential sitewide dependencies in Phase 2 is important to determining the extent of coupling between reactor units on site. The EPRI report on MUPRA (EPRI, 2021a) states that more complex and quantitative risk modeling is required to represent tight coupling between units. On the other hand, if there is loose coupling between reactors (i.e., limited or no sharing of SSCs, limited or no connected structures), then the EPRI report states that assessment of MU risk could consist of "…qualitative screening analysis and limited quantitative assessment of risk issues" that stem from sitewide dependencies. Phase 2 also is important in identifying any shared resources between the reactors and either the SFPs or the DCS facility.

The categories of potential sitewide dependencies (from Table 4-1) that are assessed in Phase 2 are:

- shared physical resources
- shared or connected systems, structures, or components (SSCs)

Although MU or sitewide IEs are already addressed in Phase 1 (see Section 4.3.1), it is recognized that analysts could identify other such IEs in the process of identifying Phase 2 (or even Phase 3) dependencies.

Separate guidance for the categories of shared physical resources and shared or connected SSCs is given below. Section 12B.6 provides more detail for the identification of Phase 2 sitewide dependencies.

4.3.2.1 Shared Physical Resources

Shared resources (e.g., electric power via common grid and/or switchyard, ultimate heat sink, water supplies for fire protection, diesel fuel, etc.) should be identified in the sitewide assessment. All the L3PRA project Level 1 and Level 2 PRA models, for all hazards, for both the “base case” and “FLEX case,” were reviewed to identify shared resources. The Level 1 PRAs were reviewed initially, followed by the Level 2 PRAs.

Some shared resources may be modeled already in the PRA via support systems. Other shared resources (e.g., water or diesel fuel supplies) may not be explicitly modeled in Level 1 PRAs but may be implied by modeling in Level 2 PRAs or in modeling FLEX strategies. Note that there can be overlap between dependencies identified in this phase with those identified in other phases of the sitewide dependency assessment. For example, electric power sources, such as the grid, would be identified in the Phase 1 sitewide dependency assessment, while specific electrical components may be identified in Phase 3 (e.g., identical electrical components in Units 1 and 2).

Also, note that adequate staffing is considered a “human or organizational dependency” rather than a physical resource in the L3PRA project categorization of sitewide dependencies. However, if concerns about adequate staffing occur to the analyst while performing the assessment of shared physical resources, notes can be made during this assessment then later transferred to the documentation of “human or organization dependencies.”

4.3.2.2 Shared or Connected SSCs

The EPRI report (2021a) provides guidance regarding the identification and modeling of shared or connected systems and components. In some cases, the single unit PRA (SUPRA) may already include some of this modeling. However, while shared systems and components may be credited in the SUPRA, only one reactor unit can credit a shared system or component in the MUPRA. While the internal events Level 1 PRA may be the source of most shared or connected systems and components, it is important to document which such systems and components are important to the results for other hazards and for the Level 2 PRA.

Shared or connected structures also may have been identified in internal flood and internal fire PRAs and Level 2 PRAs may have considered the availability and/or accessibility of equipment (and associated operator actions) for some locations inside the plant. In general, shared or connected structures should be addressed within the appropriate hazard group (e.g., internal fire or flood) and/or PRA level (e.g., Level 1 or Level 2).

Appendix B provides worksheets to help analysts to identify any potential dependencies related to shared or connected SSCs.

4.3.2.3 Assessment of “Tight” or “Loose” Coupling of Reactors

The EPRI report (2021a) states that the extent of “coupling” between two reactor units is one way to characterize MU risk. Also, an assessment of “loose coupling” can be a justification for using simplified approaches for estimating MUCDF.

From the EPRI report (EPRI, 2021a), examples of features for “loosely coupled” units are:

- limited (or no) shared systems
- major structures are separated and/or unconnected

In contrast, “tightly coupled” units have complex dependencies, examples of which include:

- shared support systems
- shared front-line systems
- inter-unit electrical dependencies
- common or shared structures

Both “loosely coupled” and “tightly coupled” units have the following types of dependencies:

- shared or common offsite power connections
- shared ultimate heat sink or cooling source
- common component types
- shared accident mitigation resources (e.g., FLEX equipment)
- common physical location
- common EOPs, operator training, etc.
- common Emergency Operations Center

The L3PRA project’s ISR task included assessment of “coupling” between the two reactors on the reference site.

4.3.3 Phase 3 Guidance for Sitewide Dependency Assessment

The Phase 3 assessment of potential sitewide dependencies addresses the remaining categories of dependencies shown in Table 4-1. The potential sitewide dependency categories assessed in Phase 3 are:

- identical components

- proximity dependencies (relevant mostly for external events)

- human and organizational dependencies

- potential accident propagation between units (may not need to be considered for loosely coupled reactors, especially if there are no shared support systems)

- potential hazards correlations (relevant for seismic events, especially)

Potential sitewide dependencies that are identified in Phase 2 are expected to be more important and are most likely to be represented with modifications to logic models. On the other hand, potential dependencies identified in Phase 3:

- are typically modeled by adjustments to basic event (BE) probabilities, rather than logic modeling

- are difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether CCF groups should be expanded and what adjustment factor to use for an expanded group)
- are difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the Technical Support Center (TSC), etc.
- typically require modeling that is beyond the PRA state of the art

As in the steps to identify sitewide IEs in the Phase 1 sitewide dependency assessment, Phase 3 potential dependencies between the two reactors are identified first. Dependencies between the SFPs and two reactors are identified next, then dependencies between dry cask storage and the two reactors.

As stated above, some of the Phase 3 categories of dependencies can be difficult to assess (e.g., due to limitations in the availability of information or the state of the art for PRA or hazard modeling). Therefore, the analysts are recommended to view the identification of such dependencies in a way similar to that for the identification of sources of uncertainty. In other words, the analyst should give their best effort to identifying such dependencies, while recognizing that not all (and, maybe, only a few) of such dependencies can be represented in risk models. Also, the analysts should not be deterred from identifying a potential dependency if such a dependency is beyond the state-of-the-art for PRA or hazard modeling. For the L3PRA project, the decision to include or represent identified sitewide dependencies was addressed in a separate ISR task (e.g., perform MUCDF calculations).

Separate guidance for each Phase 3 category of potential sitewide dependencies is given below. Section 12B.7 provides more detailed guidance for the identification of Phase 3 sitewide dependencies.

4.3.3.1 Identification of Identical Components

Identical components that are modeled in both reactor units should be considered for modeling cross-unit common cause failures (CCFs). Two different types of CCFs can be identified and documented for potential consideration in estimating MU risk:

- CCFs that are already modeled in the L3PRA project’s PRA models
- new CCFs involving identical components across the two reactors units (or between the reactors and the SFPs or dry cask storage)

For the L3PRA project, it should be noted that this identification did not include any CCFs for components that are not included in the existing L3PRA project’s PRA models. Similarly, the SFP PRA was reviewed to identify any identical components with the two reactors. Since there are no parallel systems between the SFPs and the reactors, commonalities in equipment design and function were used as the basis for this review.

Screening decisions for the L3PRA project on which CCFs are modeled in the multi-unit risk calculations were made in a later ISR task step (e.g., it may not be practical to extend all CCF groups to cross-unit CCF groups). However, analysts were asked to document any relevant screening inputs such as low risk significance (e.g., based on importance measures such as Fussell-Vesely < 0.005 or Risk Achievement Worth < 2) or limited operational experience to support calculations of CCF parameters for large group sizes.

4.3.3.2 Proximity Dependencies

Proximity dependencies arise from:

- exposure of multiple SSCs to shared phenomenological or environmental conditions
- common features between units
- operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source

Proximity dependencies may cause failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat, cold, or radiation levels), debris, explosions,[12] etc., from a nearby radiological source. External hazards may fail identical or similar structures due to common location of structures for both units. These dependencies are not likely to have been identified in individual risk models for each radiological source. External hazards and radiological concerns (e.g., for Level 2 PRA) are expected to be the principal concern for SSCs in both units that share phenomenological or environmental conditions. Dependencies related to common features between units (e.g., structures for both units are essentially in the same location or structures for both units are the same height) are likely to be important only to external hazards. Proximity dependencies involving operator actions (e.g., field operator actions taken for Unit 2 while near Unit 1) could be identified in this category but should be addressed in the category “human or organizational” dependencies (see Section 4.3.3.3 below).

Proximity dependencies for SSCs due to environmental conditions (that are not associated with external hazards) can only occur if SSCs for both reactors are shared or connected. Although shared or connected SSCs are addressed in the Phase 2 assessment of sitewide dependencies, additional assessment from the perspective of proximity dependencies should be performed. Examples of the types of hazards that involve environmental conditions that could affect SSCs due to proximity include:

- effects of fire events (e.g., heat, smoke, or toxic gases)
- radiation (for both Level 1 PRA and Level 2 PRA conditions)
- internal flooding

4.3.3.3 Human and Organizational Dependencies

It should be noted that there is limited information on how sitewide human or organizational dependencies are modeled in MUPRAs beyond the treatment of operator actions related to shared human resources or shared/connected SSCs. In addition, addressing some of these

[12] The L3PRA project has not produced any results that include the potential for explosions. However, as discussed in Section 7, potential explosions were explored in the context of a potential MU accident and Level 2 PRA.

dependencies has been recognized as being beyond the current state-of-the-art, similar to that for treatment of cross-unit CCFs.

As discussed in more detail in Section 12B.7.3 , this category has been defined differently in the EPRI report (EPRI, 2021a) and the two IAEA reports (IAEA 2019, 2021a). The definition recommended for the L3PRA project's ISR approach is intended to capture all remaining dependencies related to human and organization resources. It is also expected that some potential dependencies identified in other categories (e.g., shared physical resources or proximity dependencies) are most appropriately addressed by HRA.

As indicated in Table 4-1, the recommended definition for the "Human or Organizational" category of dependencies is:

> Dependencies between operator actions across multiple radiological sources that can result from multiple causes, including sharing of staff and shared organizational factors.

The following are examples of potential human and/or organizational dependencies discussed in the EPRI (2021a) and IAEA (2019, 2021a) reports:

- shared human resources between units
- shared control rooms
- common procedures (e.g., EOPs, AOPs, SAMGs, or FLEX procedures)
- common operator training
- common human machine interface
- common command and control structure
- common Technical Support Center (TSC)
- common Emergency Response Organization (ERO)
- common offsite support
- increased stress due to MU accident conditions
- accessibility concerns due to the other unit's degraded condition
- common environmental concerns for operators of both units (e.g., field operators taking actions at local control stations, at locations shared by both units, or outside the plant(s))

In addition, typical HRA concerns are relevant, such as:

- timing of the action (especially with respect to when conditions from one reactor can affect another reactor)
- cues and indications to prompt and/or support operator actions

- potential dependencies with prior actions

Section 12B.7.3 provides guidance on the identification of potential human and organizational dependencies in two sub-categories that are related to how they can be represented in PRA: (1) “explicit” dependencies and (2) “implicit” dependencies. Borrowing from the EPRI report (EPRI, 2021a), dependencies arising from shared physical resources and shared or common SSCs are called “explicit” dependencies. These dependencies may already be modeled or identified for representation by the Phase 2 sitewide dependency assessment. All other potential dependencies are referred to as “implicit” or “indirect” dependencies (e.g., features of shared plant contexts, such as a shared control room and TSC or common procedures and training). Implicit dependencies could be addressed using modeling assumptions or subjective judgment to adjust human error probabilities (HEPs). It should be noted that, consistent with EPRI (2021a), guidance for assessing potential implicit dependencies includes consideration of potential positive and negative influences as inputs for deciding whether to, or how to, represent potential sitewide dependencies of operator actions (e.g., between both reactors).[13]

In all cases, operator actions modeled in the single unit PRA model should be re-examined to determine if they are still feasible in the multi-unit context. Operator action feasibility criteria were developed specifically for the fire PRA context (see NUREG-1921 (NRC, 2012b) and its Supplement 1 (NRC, 2017a) and Supplement 2 (NRC, 2019)), but are applicable to other PRA hazards, including radiological concerns for Level 2 HRA. These feasibility criteria also have been adopted by the L3PRA project, as needed (e.g., for Level 2 HRA).

The feasibility factors for an operator action from the fire HRA reports (NRC, 2012b; NRC, 2017; NRC, 2019) are:

- sufficient time available for the action
- sufficient staffing for the action (i.e., “shared human resources between units” identified in Phase 2)
- primary cues available and sufficient for the action
- action is proceduralized and trained upon
- action location (and travel paths) are accessible (including consideration of environmental factors)
- needed equipment and tools are available and accessible
- relevant components are operable[14]
- action is supported by a communications plan
- action is supported by a plan for command and control

[13] It is recognized that the consideration of positive influences on operator actions is not consistent with the PRA standard (ASME/ANS, 2022). However, the L3PRA project team judged the approach in EPRI (2021a) to be appropriate for MUPRA. Note that a MUPRA standard is currently under development.

[14] For all contexts **except fire PRA**, the operability of relevant components should be addressed by the PRA model.

If any one of the above statements are not true, then the operator action should be considered infeasible.

When addressing a multi-unit or multi-source (i.e., including actions for the SFPs) accident, the analyst should consider if these feasibility criteria would be assessed differently, resulting in a different feasibility determination. In addition, some assumptions may need to be made about these factors (then re-visited as the analysis proceeds or for sensitivity analyses). For example, sufficient staffing could be assumed initially. Later in the analysis, when the timing of required actions for both units is better known, it could be determined that there is not sufficient staffing for simultaneous actions needed for both Units 1 and 2.

Examples of human or organizational dependencies (for which the probability of human error, or other type of basic event, may need to be changed to 1.0) are:

- A fire in Unit 1 MCR generates enough smoke in both MCRs to satisfy the criteria for MCR abandonment (see the fire HRA reports [NRC, 2012b; NRC, 2017; NRC, 2019]) of both units (i.e., shared structure with a common environmental hazard).
- Water resources needed to implement EDMG strategies for both reactors are only sufficient for one reactor.
- There are not enough field operators to simultaneously implement EDMG strategies for both reactors and the SFPs.
- Unit 1 reaches core damage before Unit 2 and the resulting high radiation from Unit 1 prevents the performance of a local operator action for Unit 2 (i.e., proximity of environmental hazard from Unit 1 for operator actions needed for Unit 2).

4.3.3.4 Accident Propagation

As noted in Section 12B.7.4 , there is limited guidance on this topic at present. EPRI (2021a) recommends that the results of the Phase 2 assessment of coupling between the two reactors should be performed first. If the Phase 2 assessment is that the reactors are “loosely coupled,” then this assessment of accident propagation can be limited for Level 1 PRA. However, this type of dependency may need to be re-visited when considering Level 2 PRA.

Several places in EPRI (2021a) reference the potential for “cascading” failures with respect to initiating events that involve shared support systems. However, if there are no shared support systems, then those types of dependencies would not be relevant. EPRI (2021a) uses the phrase “propagating failures,” sometimes interchangeably with “cascading failures,” but also to refer to internal fires and internal flooding events.

4.3.3.5 Hazards Correlations

This category of dependencies addresses SSCs and operator actions that may be affected in the same or similar ways by various hazards. For example, potential hazards correlations are especially relevant for seismic events (but may also be relevant for other external hazards). Such dependencies are likely to be already addressed in the base PRA for each unit. However, coincident failures of operator actions and/or SSCs for each radiological source need to be addressed in calculating MU and multi-source risk.

The analyst should use judgment and knowledge of the current state-of-the-art to select and represent multi-unit and sitewide hazard correlations for both MU Level 1 and MU Level 2 PRA. Examples of papers consulted by the project team for the assignment of hazard correlations for MU Level 1 PRA include: Abrahamson (1993), Kawakami (2003), Zerva (2009), and DeJesus-Segarra (2020).

4.4 Results for Phase 1 Sitewide Dependency Assessment

This section summarizes the results of the Phase 1 sitewide dependency assessment. These results consist of identified multi-unit initiating events (MUIEs) and sitewide initiating events (IEs). Appendix C provides the complete set of results for the Phase 1 sitewide dependency assessment.

Table 4-2 summarizes the L3PRA project single unit CDF results for all modeled hazard categories. Internal events (42.4%) and internal fires (40.7%) are the largest contributors to single unit CDF, with seismic events (7.2%) and high winds (9.2%) also being significant contributors.

Table 4-3 combines the results from all the Phase 1 sitewide dependency results tables given in Appendix C. This table displays the results from applying the converse of the criteria provided in Section 4.3.1. If ***any*** of the “converse criteria” were satisfied, then the potential MUIE was retained for consideration in the L3PRA’s ISR task. The “converse criteria” are:

1. The IE immediately results in reactor trip in both units.

2. The IE immediately results in reactor trip of one unit and a degraded condition in the second unit.

3. The IE immediately results in degraded conditions in both reactor units.

The advantage of using the “converse criteria” rather than the originally defined criteria is that there are many fewer IEs that satisfy the converse criteria rather than the original criteria for the reference site. In addition, the converse criteria results (i.e., identified potential MUIEs) are directly useful to later ISR tasks.

Table 4-3 shows all the IEs, for all hazards, that were identified as potential MUIEs or sitewide IEs. The table is organized as follows:

- Results for reactors are shown in three columns:
    - Results of applying the converse criteria (e.g., “yes” the IE should be retained as a potential MUIE)
    - Clarifying notes or notes with recommendations or caveats to the application of the converse criteria
    - CDF results from the single unit PRA – both absolute frequency and percentage of the total CDF for the applicable hazard category (e.g., the CDF for grid-related losses of offsite power is 1.8×10^{-5} per reactor-critical-year, which is 29 percent of the total CDF for internal events) (see the explanation of the Table 4-3 organization by rows below)

- Results for SFPs and DCS together are shown in four columns:

    - Relevance (yes or no) of each initiator to SFPs or DCS, since the original and converse criteria do not apply to the SFPs and DCS

    - Two columns show risk results, if there are any, for SPFs and DCS:

        - SFP results shown in purple shaded rows
        - DCS results shown in blue shaded rows

    - Clarifying notes or notes with recommendations or caveats

- Results for different hazards are shown in sections using green-shaded row headings. These row headings are:

    - internal events
    - internal floods
    - internal fires
    - seismic events
    - high winds
    - low power and shutdown conditions (as applied to the SFPs only)

The risk metrics used to determine the risk significance of IEs and hazards are CDF for the reactor and significant fuel uncovery frequency (SFUF) for the spent fuel pools (which is analogous to CDF). The blue and purple shading used in Table 4-3 shows that the only at-power IEs that are important for the SFPs and DCS are certain seismic events.

In summary, Table 4-3 shows that:

- The following potential MUIEs are important to the reactors only:

    - LOOPs

    - fire events/scenarios given in Table 4-3

- Seismic events are important to the reactors, SFPs, and DCS

    - all bins are important to the reactors and the SFPs

    - Bins 1-6 are the most important to the reactors

    - Bins 5-7 are most important to the SFPs (with bin 7 having the largest contribution to risk)[15]

    - Bins 5-7 are important to DCS

[15] Seismic bins 1, 2, 3, and 8 make very small contributions to overall spent fuel uncovery frequencies. For bin 8, the seismic initiating event frequency is low. For bins 1, 2 & 3, the probabilities of SFP failures (e.g., liner failures) are low because the SFP is robustly built. Also, the amount of sloshing for bins 1, 2, and 3 was determined to be insignificant.

Note that only at-power conditions were addressed by the ISR task. As the last line in Table 4-3 states, a low-power and shutdown (LPSD) PRA was performed for internal events only. Because LPSD results for other hazards are not available, LPSD conditions were not addressed by the ISR task.

The results shown in Table 4-3 were used as inputs to decisions made for later steps in the ISR task, such as which sitewide IEs are represented in multi-source risk calculations. Other inputs (e.g., results of the Phase 2 sitewide dependency assessment) were also used in this decision-making process.

Table 4-2 Summary of CDF Results from Level 1 PRAs for Single Reactor

Hazard	CDF (/rcy*)	Percentage of Total CDF
Internal events	6.39E-5	42.4%
Internal floods	7.91E-7	0.5%
Internal fires	6.14E-5	40.7%
Seismic events	1.08E-5	7.2%
High winds	1.38E-5	9.2%
Total Single Unit CDF	**1.51E-4**	**100%**

*rcy – reactor-critical-year

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards

Reactors				SFPs and Dry Cask Storage			
Potential MUIE	Converse Criteria Met? (Yes or No)	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ Dry Cask Storage? (Yes or No)	Risk Metric	% of Total⁺	Refinement/Caveat Notes
Internal Events							
Grid-Related Loss of Offsite Power (LOOP)	Yes (#1)	Sitewide LOOP would occur.	1.8E-5 (29%)	No	Exact contribution is unknown; sensitivity study for SFPs suggests that contribution would be small.		Screened out of base case SFP and dry cask storage analyses.
Switchyard-Centered LOOP	Yes (#1)	Could result in sitewide or single unit LOOP.	1.0E-5 (16%)	No			
Weather-Related LOOP	Yes (#1)	Likely sitewide LOOP, but not definite.	9.0E-6 (14%)	No			
Loss of Nuclear Service Cooling Water (NSCW)	No	If cross-unit CCF is considered, dual-unit loss of NSCW can occur. The dominant loss of NSCW cutsets are from pump CCF. This scenario is not recommended to be screened out.	8.8E-6 (14%)	No	Unknown; see above.		Screened out of base case SFP analysis. N/A for dry cask storage.

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and Dry Cask Storage			
Potential MUIE	Converse Criteria Met? (Yes or No)	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ Dry Cask Storage? (Yes or No)	Risk Metric	% of Total+	Refinement/Caveat Notes
Interfacing System LOCA (ISLOCA) from Residual Heat Removal (RHR) Hot Leg Suction Lines#	No	If cross-unit CCF of the RHR hot-leg suction isolation valves is considered, dual-unit ISLOCA can occur. This scenario is not recommended to be screened out - dominant CCF aspects and high-risk potential of dual unit ISLOCA.	2.3E-7 (<1%)	N/A			
ISLOCA from RHR Cold Leg Injection Lines [Two IEs]#	No	If cross-unit CCF of the RHR cold-leg injection isolation valves is considered, dual-unit ISLOCA can occur. This scenario is not recommended to be screened out - dominant CCF aspects and high-risk potential of dual unit ISLOCA.	8.4E-8 (<1%)	N/A			
Internal Floods							
1-FLI-TB_500_HI1	Yes; #3 possible	Flood in turbine building, main condenser	Not significant	N/A			
1-FLI-TB_500_LF	Yes; #3 possible	Flood in turbine building, circulating water expansion joint failure	1.6E-8 (2.1%)				
1-FLI-TB_500_LF-CDS	Yes; #3 possible	Flood in turbine building, piping failure	Not significant				

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and Dry Cask Storage			
Potential MUIE	**Converse Criteria Met? (Yes or No)**	**Refinement/Caveat Notes**	**CDF (/rcy) (%)***	**Relevant to SFPs/ Dry Cask Storage? (Yes or No)**	**Risk Metric**	**% of Total+**	**Refinement/Caveat Notes**
1-FLI-TB_500_HI2	Yes; #3 possible	Flood in turbine building, main condenser	Not significant				
Internal Fires							
MU-IE-FRI-1	Yes; #1 and #2	Both MCRs evacuated (CCDP = 1); MCR abandonment scenarios	0.2%	N/A			MCR evacuation scenarios contribute less than 1% to CDF from internal fire events, and consequently, even less to the total plant CDF. However, with a 1.4E-07/rcy MUCDF (their CCDP is 1.0), they should be retained.
MU-IE-FRI-2		Scenarios with shared areas “A+Y” between Units 1 and 2	16.3%				These fire scenarios are mapped together, and a representative scenario should be defined and further evaluated.
MU-IE-FRI-3		Unit 1 fires that cascade to Unit 2	68.9%				These fire scenarios are mapped together, and a representative scenario should be defined and further evaluated.
MU-IE-FRI-4		Unit 2 fires that cascade to Unit 1	5.4%				These fire scenarios are mapped together, and a representative scenario should be defined and further evaluated.

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and Dry Cask Storage			
Potential MUIE	Converse Criteria Met? (Yes or No)	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ Dry Cask Storage? (Yes or No)	Risk Metric	% of Total+	Refinement/Caveat Notes
Seismic Events+							
MU-IE-EQK-1	Yes; #1 and #2	Seismic event in bin 1 (0.1–0.3G) occurs	12.0%	Yes (SFPs only)	SFUF**	0.0%	Negligible contribution to SFUF
MU-IE-EQK-2		Seismic event in bin 2 (0.3–0.5G) occurs	11.3%				
MU-IE-EQK-3		Seismic event in bin 3 (0.5–0.7G) occurs	15.0%	Yes (SFPs only)	SFUF**	0.9%	Small contribution to SFUF
MU-IE-EQK-4		Seismic event in bin 4 (0.7–0.9G) occurs	22.5%	Yes (SFPs only)	SFUF**	5.1%	
MU-IE-EQK-5		Seismic event in bin 5 (0.9–1.1G) occurs	20.8%	Yes (SFPs)	SFUF**	15.5%	
				Yes (DCS)	LCF risk 0-10 miles	See Notes	Two potential consequences: (a) failing auxiliary building during cask loading, and (b) tipping and failing casks on the pad. Total for (a), Bins 5-7: 0.01%. Total for (b), Bins 5-7: 0.00%. Very small source term.
MU-IE-EQK-6		Seismic event in bin 6 (1.1–1.5G) occurs	16.2%	Yes (SFPs)	SFUF**	37.6%	
				Yes (DCS)	LCF risk 0-10 miles	See Notes	See bin 5
MU-IE-EQK-7		Seismic event in bin 7 (1.5–2.5G) occurs	2.2%	Yes (SFPs)	SFUF**	40.5%	
				Yes (DCS)	LCF risk 0-10 miles	See Notes	See bin 5
MU-IE-EQK-8		Seismic event in bin 8 (2.5G and above) occurs	0.02%	Yes (mostly SFPs)	SFUF**	0.4%	Small contribution to SFUF; even smaller contribution for dry cask storage risk.

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and Dry Cask Storage			
Potential MUIE	**Converse Criteria Met? (Yes or No)**	**Refinement/Caveat Notes**	**CDF (/rcy) (%)***	**Relevant to SFPs/ Dry Cask Storage? (Yes or No)**	**Risk Metric**	**% of Total⁺**	**Refinement/Caveat Notes**
High Winds							
MU-IE-WIND-1	Yes; #1 and #2	SBO and wind damage to SSCs	100%	N/A			All wind scenarios modeled for Unit 1 are mapped into this scenario. A representative MU scenario can be assigned to this scenario. (If wind scenarios were to be considered individually, they could have been inadvertently screened out. Together, all wind scenarios contribute only 5% to the total plant CDF. They are mostly loss of offsite power events, with insignificant damage to safety-related SSCs, even at high wind speeds.)
LPSD conditions, SFP analysis							
Non-seismic LLOINV		LPSD PRA was performed for internal events only. Because no other hazards were addressed by LPSD PRA, LPSD was not addressed by the ISR task.		Yes (SFPs only)	SFUF**	0.0%	Applicable when one unit is shut down for refueling and it is connected to the SFP.

(See the following page for table notes.)

Table 4-3 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

* *Percentage of CDF for that specific hazard.*

\# *Note that conditional failure probabilities were treated through expert elicitation.*

\+ *Percentage of SFUF from all hazards.*

** *SFUF: Significant Fuel Uncovery Frequency (analogous to CDF)*

4.5 Results for Phase 2 Sitewide Dependency Assessment

This section summarizes the results of the Phase 2 sitewide dependency assessment. There are three types of results for this sitewide dependency assessment:

1. shared physical resources
2. shared or connected SSCs
3. assessment of coupling between the two reactor units

The results provided in this section address the two reactors, the SFPs, and the DCS facility. The "base case" results correspond to the overall freeze date for the L3PRA project of August 2012 (with a few exceptions). However, as noted previously, sensitivity analyses for FLEX strategies have been performed for the two reactors. A similar sensitivity analysis for FLEX strategies was not performed for the SFPs.

The Phase 2 sitewide dependency assessment involved several assessments that were performed in succession by different analysts, with each analyst building on the previous assessment. The order of inputted results from the analysts was:

1. Level 1 PRA for internal events for the two reactors

2. Level 1 PRA for internal floods for the two reactors

3. Level 1 PRAs for fire, seismic, and wind for the two reactors

4. FLEX strategies for the two reactors

5. Level 2 PRA for all hazards for the two reactors

6. Level 1 and 2 PRAs for all hazards for the SFPs [16] and the DCS facility

Appendix D provides more detailed results for the Phase 2 sitewide dependency assessment.

4.5.1 Results for Shared Physical Resources

Results for shared physical resources are given below for the two reactors, the DCS facility with the two reactors, and the SFPs with the two reactors. Appendix D provides additional details for the Phase 2 sitewide dependency assessment.

In summary, Table 4-4 shows that there are only three physical resources shared between the two reactors. Two are related to electric power needs: (1) 230 kV and 500 kV switchyards, and (2) the alternate switchyard. The main switchyards (and offsite power sources) were identified in the Phase 1 identification of sitewide IEs. Consequently, this dependency was addressed in the MU risk model as a sitewide IE. The alternate switchyard, on the other hand, can be used to supply power to only one of the two units (and is currently credited in the single unit PRA model). So, addressing this dependency required modeling an asymmetry between the two reactor units (i.e., only one unit can credit use of the alternate switchyard) for relevant IEs.

[16] Note that the two SFPs are treated as a single large pool in the L3PRA project because they are hydraulically connected for most plant operating states.

The third shared resource between the two reactors is water, namely the water contained in the tanks that are used with B.5.b pumps in EDMG strategies in response to Level 2 PRA scenarios. At present, the needed volume of water for success of such EDMG strategies is assumed to be equivalent to both fire water storage tanks (FWSTs). However, the smaller volume demineralized water storage tank (DWST) is indicated to be an option, too. It is not currently known whether EDMG strategies can be successful with the smaller volume DWST. Also, it is not known if other water sources are available (and what procedures, training, etc., would support their use).[17]

Based on review of the relevant L3PRA project PRAs, there are no shared physical resources between DCS and the two reactors. The DCS facility is a separate facility that does not require any external resources (e.g., electric power or cooling water) to prevent fuel damage. Passive design of the casks and the facility are sufficient to maintain necessary cooling and fuel configuration, even in the case of the most damaging seismic event considered in the L3PRA project.[18]

Table 4-5 shows the potential dependencies between the SFPs and the two reactors for both the SFP PRA base case (i.e., significant fuel uncovery within 7 days after accident initiation) and sensitivity case (i.e., significant fuel uncovery within 14 days after accident initiation) performed for the SFP Level 1 and Level 2 PRAs. Both the base analysis and the sensitivity analysis credit the same two strategies (internal and external) from the EDMGs. Both of these strategies, in turn, have multiple options for restoring level for the SFPs (e.g., multiple locations for standpipe valves) and use two different approaches (i.e., makeup or spray).

For the base case SFP PRA, there is sharing of the following physical resources between the SFPs and the reactors:

- electric power sources (i.e., switchyards)
- ultimate heat sink, NSCW basins, and NSCW intake structures
- water tanks outside plant buildings
    - FWSTs
    - DWST
- water supplies for refilling water tanks

There are similar shared physical resources between the SFPs and the reactors for the sensitivity case of the SFP PRA:

- electric power sources (i.e., the grid via the switchyards)
- ultimate heat sink, NSCW basins, and NSCW intake structures
- water in tanks inside the plant, such as:
    - refueling water storage tanks (RWSTs)
    - reactor makeup water storage tanks (RMWSTs)
    - DWST

[17] The reference plant Technical Support Guideline has a table for “Water Sources” but the DWST is not included.

[18] The exception to these statements is for the very short amount of time during cask loading where SFP water is circulated through the cask. However, there are several backup strategies for restoration of cooling, including returning the cask to the SFP.

It should be noted that FLEX strategies have not been addressed for the SFPs. In particular, a FLEX sensitivity case for the SFP Level 1 and Level 2 PRAs (similar to that which was done for the reactor Level 1 PRAs) was not developed as part of the L3PRA project.

Table 4-4 Shared Physical Resources Between the Two Reactor Units

Identified Dependencies	Relevant IEs and MUIEs	Relevant Hazards	Notes	Key Inputs for Modeling Decisions
230kV and 500kV Switchyards	All LOOPs	Internal events, internal fires, external events (e.g., seismic events or high wind events)	The reference plant electrical system notebook states that there are both 230kV and 500 kV switchyards, but that these two switchyards are connected through two 230/500 kV auto-transformers. The two independent offsite power sources are separated physically as they leave the 230kV substation and are arranged so that no one event such as a falling line, tower, or other structure will damage both lines. This statement is in the electrical system notebook, "*Since no major equipment, electrical buses, or EDGs are shared between Units 1 and 2, the impact on either of a loss of offsite power occurring simultaneously at both units can be analyzed by two independent Unit 1 and 2 models*." Under normal operation Unit 1 Division I and Unit 2 Division II are fed by one offsite source, while Unit 1 Division II and Unit 2 Division I are fed from the other offsite source.	These dependencies also were captured in Phase 1, identification of sitewide IEs. These dependencies will be addressed via sitewide IEs.
Alternate Switchyard	Plant-centered, switchyard, and consequential LOOPs	Internal events	Can only supply one unit at a time. The alternate switchyard is already assumed to be unavailable for weather- and grid-related LOOPs. May have limited impact since plant and switchyard LOOPs are less likely to be MUIEs.	This is an important dependency that can only be captured in development of the multi-unit PRA model. Likely, Unit 1 will be credited with use of the alternate switchyard, and Unit 2 will not. This results in an asymmetry between the two reactor units.

Table 4-4 Shared Physical Resources Between the Two Reactor Units (cont.)

Identified Dependencies	Relevant IEs and MUIEs	Relevant Hazards	Notes	Key Inputs for Modeling Decisions
North and South Fire Water Storage Tanks (FWSTs)	Level 2 scenarios	Internal events and internal floods, external events (e.g., seismic events)	The Level 2 PRA report (NRC, 2022b) describes the equipment and resources needed to implement Extensive Damage Mitigation Guidelines (EDMGs) in response to post-core-damage scenarios.	The volume from **both** FWSTs (total of 600,000 gallons) is used to implement the associated EDMG strategies. The demineralized water storage tank (DWST) can be used as a water source; however, the DWST has a smaller volume.

Table 4-5 Shared Physical Resources Between the SFPs with the Reactors

Identified Dependencies	Rx MUIEs	Relevant Hazards	Modeling or Screening Notes
Level 1 and 2 PRAs – Base and Sensitivity Analyses			
230kV and 500kV Switchyards	All LOOPs	Internal events, seismic events	*Internal EDMG strategy:* Specifically, electric power is needed to operate the NSCW systems in order to replenish SFP inventory. *Sensitivity case only:* Offsite power is used to facilitate normal cooling of SFPs via NSCW standpipes.
Ultimate heat sink and associated intake structure	All LOOPs	Internal events, seismic events	*Internal EDMG strategy:* Specifically, the water inventory in the NSCW systems is needed to replenish SFP inventory via NSCW standpipes.
Water storage tanks: FWSTs (2) and DWST (1)	All LOOPs	Internal events, seismic events	*External EDMG strategy:* Specifically, the water inventory in the FWSTs or DWST is needed to replenish SFP inventory using a B.5.b pump.
Water supply for refilling FWSTs and DWST	All LOOPs	Internal events, seismic events	*External EDMG strategy:* Specifically, the water inventory in the FWSTs or DWST may need to be replenished.
Various water tanks inside plant buildings (e.g., RWSTs, RMWSTs, or DWST)	All LOOPs	Internal events, seismic events	*Sensitivity case only, as documented in the Level 1 and Level 2 SFP PRA report (NRC, 2025a): These tanks are used for the gravity-feed strategy.*
FLEX Strategies			
FLEX pumps and associated equipment	All LOOPs	Internal events, seismic events	*FLEX not formally addressed in the L3PRA project's SFP PRAs.*
Various water sources	All LOOPs	Internal events, seismic events	*FLEX not formally addressed in the L3PRA project's SFP PRAs.*

4.5.2 Results for Shared or Connected SSCs

This section summarizes the results of the Phase 2 sitewide dependency assessment for shared or connected SSCs. Results for the two reactors are given first, followed by the results for the SFPs with the two reactors. There are no shared or connected SSCs for the DCS facility with either the two reactors or the SFPs. Appendix D provides more detailed results.

Table D-3 provides the full results of shared or connected SSCs for the two reactors. In summary, Table D-3 shows that:

- The only common systems or components between the two reactors are the B.5.b pumps and associated equipment needed for Level 2 PRA scenarios.

- The only common or shared structure between the two reactor units is the FLEX building.[19] However, since FLEX buildings have been specifically designed and constructed to withstand external events, failure of the FLEX building is not considered in the ISR task for either external or internal events. Also, per the reference plant Final Integrated Plan (FIP), the FLEX building is seismically qualified and "[l]arge portable FLEX equipment such as pumps and power supplies are secured, as required, inside the FLEX Storage Building to protect them during a seismic event…"

- There are several buildings that are connected between the two units: (a) auxiliary buildings, (b) control buildings (including the Technical Support Center (TSC)), (c) main control rooms (MCRs), (d) cable spreading rooms, and (e) turbine buildings.

Table D-3 also provides some important notes on the shared or connected buildings between the two reactors, such as:

- None of the building connections are considered important dependencies for internal events and internal floods PRAs.

- All the building connections are flagged as being potentially important for seismic events but are considered to be best addressed in Phase 3 of the sitewide dependency assessment.

- The connection between the MCRs of Units 1 and 2 is an important dependency for certain fires that could produce enough smoke to prompt abandoning both MCRs.

- Connections between the auxiliary buildings, control buildings, and turbine buildings are identified as being potentially important dependencies for fire events. There are multiple scenarios in the single unit, base fire PRA for which a fire in Unit 2 propagates and leads to core damage in Unit 1. The specific fire locations and associated equipment and connections for these scenarios is not well-understood at this time due to limited available documentation of the reference plant fire PRA.

[19] Although the reactors share the fuel handling building, it is not noted here since the SFPs are considered a separate radiological source in this dependency assessment.

Table D-4 provides the full results of the Phase 2 sitewide dependency assessment for shared or connected SSCs between the SFPs and the two reactors for all hazards for both the Level 1 and 2 PRAs. Both the SFP base and sensitivity cases (i.e., significant fuel uncovery within 7 days and 14 days after accident initiation, respectively) are shown. In particular, Table D-4 shows the SFPs share the following systems and components with the two reactors:

- the NSCW systems (internal EDMG strategy – SFP base case), including:
    - NSCW pumps
    - NSCW tower fans
- the fire protection system (specifically, standpipes and hoses) (internal EDMG strategy – SFP base case)
- B.5.b pump (external EDMG strategy – SFP base case)
- EDGs (SFP sensitivity case)
- valves needed to facilitate gravity makeup from the RWST, RMWST, or DWST (SFP sensitivity case)

Similarly, the SFPs share the following structures with the two reactors for both the SFP base analysis and sensitivity analysis:

- auxiliary building
- fuel handling building
- NSCW intake structure

In summary, for SFPs and the reactors, Table D-4 shows that:

- For the SFP base case, there is a clear dependency due to sharing of equipment and personnel via EDMG strategies.
- For the SFP sensitivity analysis, there also are clear, though fewer, dependencies for the EDMG strategy that uses the SFPCPS.
- The SFPs share three structures with the two reactor units (i.e., auxiliary building, fuel handling building, and NSCW intake structure).

4.5.3 Results for the Assessment of Coupling Between the Two Reactors

Using the results discussed above for the Phase 2 sitewide dependency assessments, the reactors on the reference site were assessed for “coupling” per the guidance provided in Section 4.3.2.3 and Appendix B. Table 4-6 summarizes the results of the assessments for shared physical resources and shared or connected SSCs with respect to the characteristics of “tight” coupling. From Table 4-6, it can be seen that the only assessments of potential “tight” coupling between the reactors are for certain fire scenarios and for seismic events (which are addressed in the Phase 3 assessment of sitewide dependencies, as discussed below).

Consequently, the two reactors are considered to be “loosely coupled” for all hazards except certain fire scenarios and seismic events.

If there is “loose” coupling between reactors, EPRI (2021a) states that assessment of MU risk could consist of “…qualitative screening analysis and limited quantitative assessment of risk issues” that stem from sitewide dependencies. The L3PRA project’s ISR task also used this assessment as partial justification for the simplified approach for estimating MUCDF.

Table 4-6 Summary of the Reference Site’s Features Associated with “Tightly Coupled” Reactors

Potential dependencies	Tightly Coupled? (yes or no)	Notes
Shared support systems	**No**	The two reactors do not share any support systems.
Shared front-line systems	**No**	The two reactors do not share any front-line systems.
Shared components	**No**	
Inter-unit electrical dependencies	“**Yes**” for shared switchyards and alternate power source	This dependency is common to both “tightly” and “loosely” coupled reactors.
Shared physical resources	**“Yes”** for FWSTs needed to implement EDMG strategies in Level 2 PRA.	Relevant for multi-unit Level 2 PRA only; will need to account for this in multi-unit model.
Common or shared structures	Internal events and internal floods: Assessment is “**No**” for the auxiliary buildings, control buildings, and turbine buildings.	Although these buildings are connected, equipment is not close by.
	Internal fires: Assessment is “**Yes**” for certain scenarios involving shared or adjacent spaces and for main control room abandonment scenarios.	Both units share auxiliary building, control building, fuel handling building, and turbine building. Additionally, there are other areas, such as low and high voltage yards containing equipment from both units. It should be pointed out that SSCs for redundant trains and trains from different units do not coexist in the same fire zone. Also, main control room abandonment needs to be represented in MU risk models.
	Seismic events and other external hazards: Assessment is “**Yes**” for all common and connected buildings.	This type of dependency is addressed in the Phase 3 sitewide dependency assessment and needs to be addressed in MU risk models.

4.6 Results for Phase 3 Sitewide Dependency Assessment

This section summarizes the results for the Phase 3 sitewide dependency assessments. Appendix E, Appendix F, and Appendix G provide details on these assessments.

As a reminder, Section 4.3.3 also states that all potential dependencies identified in the Phase 3 assessment are:

- typically modeled by adjustments to basic event probabilities, rather than logic modeling

- difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether CCF groups should be expanded and what adjustment factor to use for an expanded group)

- difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the Technical Support Center, etc.

- typically require modeling that is beyond the PRA state of the art

4.6.1 Results for the Identification of identical Components

This section summarizes the results of the identification of identical components that could be modeled as sitewide dependencies. Results for the two reactors and the SFPs are given separately. There are no identical components between the DCS facility and the two reactors. As stated in Section 4.3, this analysis did not address:

- combinations of only one reactor unit with either the SFPs or dry cask storage
- any combinations that do not involve both reactors
- any plant operating states beyond at-power operations

Appendix E provides detailed results for this sitewide dependency assessment.

4.6.1.1 Identical Components for the Two Reactors

Based on the approach described in Section 4.3.3.1 (and in Section 12B.7.1), identical components that are modeled in both reactor units were considered for modeling cross-unit (or inter-unit, or multi-unit) common cause failures (CCFs). Two different types of CCFs were identified and documented for potential consideration in the ISR task:

1. CCFs that are already modeled in the L3PRA project's PRA models
2. new CCFs involving a single identical component in each of the two reactor units

Note, for the first type of CCF (i.e., multiple, identical components within a single unit), no new CCFs were identified (i.e., only CCFs already included in the existing L3PRA project's PRA models were addressed).

Level 1 and 2 PRAs for internal events, internal floods, internal fires, wind-related events, and seismic events were reviewed to identify CCFs that are already modeled and could be modeled as inter-unit (or multi-unit) CCFs with a group expansion. Appendix E provides the detailed results of this review, which can be summarized as follows:

- For internal events, Table 4-7 summarizes the systems, components, and failure modes involved in modeled CCFs. This table shows that there are many potential CCFs that could be modeled as cross-unit CCFs with expanded group sizes. Bold font is used in

Table 4-7 to indicate which components and failure modes are most risk-significant per risk importance measures calculated in the internal events PRA (e.g., the importance measures typically used in SAPHIRE of Fussel-Vesely greater than 0.005 and Risk Achievement Worth greater than 2), in order to limit the number of cross-unit CCFs that need to be modeled.

- For internal floods, only one system and associated component and failure mode was considered risk-significant (i.e., containment isolation valves fail to operate).
- For internal fires and external hazards (i.e., wind-related events and seismic events), all possible systems, components, and failure modes are identified with results similar to those for the internal event results.
- Overall, there appears to be some overlap of CCFs for the same systems, components, and failure modes between different PRAs and hazards. This should be considered when deciding on the approach for modeling cross-unit CCFs in the L3PRA project's MU risk calculations.

Consideration of risk-significance of CCF groups is used in later ISR task steps because the number of potential cross-unit CCFs is too large for all to be included in the MU risk model or when performing MU risk calculations.

Regarding new potential, inter-unit CCFs, most of these potential CCFs were identified from the Level 1 PRA for internal events, and none from Level 1 PRAs for internal fires, wind-related events, and seismic events. The following single components modeled in the reactor PRAs were identified as candidates to be modeled as cross-unit (or inter-unit) CCFs in the MU risk model:

- AFW – turbine-driven pumps fail to run
- electrical – DC buses
- ECCS – NCPs fail to run
- B.5.b pumps (Level 2 PRA EDMG strategy)
- firewater storage tanks (Level 2 PRA EDMG strategy)
- 480 V FLEX DGs
- SG FLEX pumps
- boron injection FLEX pumps
- RCS makeup FLEX pumps
- FLEX fuel tankers
- FLEX tow vehicles
- makeup FLEX pumps

It should be noted that there are more than two identical components for some FLEX equipment, although no FLEX equipment CCFs were modeled in the FLEX sensitivity case.[20] One potential strategy for modeling CCFs for the MU risk model would be to model new cross-unit CCFs with the appropriate group size.

[20] For the L3PRA project, the FLEX sensitivity case involved a simplified treatment where the failure probabilities used for FLEX and manual TDAFW pump operations are parametric values chosen by expert judgment and incorporate the probability of operator or equipment failure into a single failure probability. As such, there is no value in developing CCF groups for FLEX equipment for use in the MU risk model for the L3PRA project.

Table 4-7 Level 1 PRA for Internal Events Intersystem CCF List

System	Components (*ordered by risk importance*)
Nuclear Service Cooling Water (NSCW)	**Pumps (FTR)**, cooling tower (CT) spray valves (FTO, FTC), pumps (FTS), pump motor-operated valves (MOVs), CT fans (FTS, FTR)
Switchyard	**Reserve Auxiliary Transformer (RAT) Breakers (FTO)**
Emergency diesel generators (EDGs)	**Load Sequencers, EDGs (FTR/FTS), fuel oil transfer pumps (FTS**, relays, FTR**)**, vent dampers, vent fans, running relays
Auxiliary Feedwater (AFW)	**Pumps (FTR)**, pump check valves (suction and discharge), feedline check valves, control valves, minimum flow valves (transmitters)
Electrical	Battery chargers, inverters
Reactor Protection System (RPS)	Rod cluster control assemblies, reactor trip breakers, bistables, analog process logic modules, undervoltage drivers, solid state logic
Instrumentation and Control (I&C)	ESFAS
Emergency Core Cooling System (ECCS)	Residual heat removal (RHR) pumps (FTS, FTR), RHR pump discharge check valves, containment sump suction and check valves, containment sumps, safety injection (SI) pump minimum flow valves, refueling water storage tank (RWST) suction valves (FTC), high pressure recirculation (HPR) suction check valves, high pressure injection (HPI) and low pressure injection (LPI) cold leg (CL) suction check valves, SI pump suction from RHR pumps valves, normal charging valves (FTC), centrifugal charging pumps (CCPs) (FTS)
Auxiliary Component Cooling Water	Pumps (FTR)

4.6.1.2 Identical Components Between the SFPs and the Two Reactors

A preliminary identification of identical components between the SFPs and the two reactors was done and did not find any such common components. Since the credited mitigation strategies for the SFPs mostly involve portable equipment (e.g., B.5.b pumps) that were addressed under the category of “shared or connected SSCs,” there were very few active types of equipment to review. Some of these strategies involved use of fire protection piping and valves, but these valves are different in design and function than those used in safety-related reactor systems.

Due to project scope limitations, the FLEX case for SFPs was not performed. However, the reference site FIP for implementation of FLEX strategies was used to identify how the following FLEX equipment could be modeled with new CCF basic events:

- SFP FLEX submersible pump hydraulic units
- SFP FLEX pump submersible pumps
- sets of monitor spray nozzles for SFP spray and connection equipment

4.6.2 Results for the Identification of Human and Organizational Sitewide Dependencies

This section summarizes the results for the identification of human and organizational dependencies. Appendix F provides more detailed results. Following the guidance for identifying these sitewide dependencies provided in Section 4.3.3.3 and Section 12B.7.3 , results for the identification of potential human and organizational dependencies are given for the two sub-categories of “explicit” and “implicit” dependencies.

4.6.2.1 Explicit Human and Organizational Dependencies

As stated in Sections 4.3.3.3 and 12B.7.3 , “explicit” dependencies were expected to either already be modeled or identified for representation by the Phase 2 sitewide dependency assessment. However, results from all the other sitewide dependency assessments were reviewed to develop these results.

In summary, the potential “explicit” human and organizational dependencies that were identified are, by source of information:

- From the Phase 1 sitewide dependency results, there is one potential sitewide IE that could merit attention for potential human and organizational dependencies: loss of NSCW. For the L3PRA project, the frequency of this IE was determined using fault tree (FT) modeling, especially considering different combinations of CCFs for the NSCW system. In addition, since only two of the six NSCW pumps are normally running, operator actions to start additional pumps is included in the FT for this IE. In principle, common factors could result in failed operator actions for the NSCW system that affect both reactor units (and, for some cases considered by the SFPs). It could be argued that this modeling is an “explicit” human and organizational dependency.

- From the Phase 2 sitewide dependency assessment for shared physical resources, there are potential implications with respect to HFE modeling and cross-unit dependencies for the following:

- Switchyards: For example, the same operator actions taken to restore switchyard-related losses of offsite power (LOOPs) for one reactor also restores power for the second reactor. ***Such operator actions should be modeled as single actions that affect both units.***

- The alternate switchyard: Only one reactor can be connected to the alternate switchyard. So, for certain LOOPs, the second reactor would not have offsite power while the first reactor would.

- FWSTs: The FWSTs are called out for use when implementing the Extensive Damage Mitigation Guidelines (EDMGs), as noted in the Level 2 PRA report (NRC, 2022b). However, the Level 2 HRA defines "success" as the use of both FWSTs for a single reactor. Consequently, the FWSTs can be used for only one of the two reactors and the operator action (and associated EDMG strategy) is no longer feasible for the second reactor if the FWSTs have been used for the first reactor.[21] For example, according to the Phase 2 sitewide dependency assessment results documented in Appendix D, "*the needed volume of water for success of such EDMG strategies is assumed to be equivalent to both FWSTs. However, the smaller volume DWST is indicated to be an option, too. It is not currently known whether EDMG strategies can be successful with the smaller volume DWST.*"

- From the Phase 2 sitewide dependency assessment of shared or connected structures:

    - Unit 1 and Unit 2 share the control building although there is some separation by walls and doors. Although this sharing is not expected to be an important dependency for internal events, these connections were later considered relevant for MU fire PRA.[22]

    - The Technical Support Center (TSC) is common to both units, is located in the control building, and has the same dependency assessment as for the control building.

    - The fuel handling building is common to Units 1 and 2 and houses both spent fuel pools. Because the SFPs are considered a separate radiological source in the ISR task, the fuel handling building is not considered a shared structure for this analysis.

- From the Phase 2 sitewide dependency assessment of shared or connected components for the base Level 1 PRA studies, the only type of shared components that are identified as important to human and organizational dependencies are the B.5.b pump and its associated equipment. While there are two B.5.b pumps (and associated equipment) to implement EDMG strategies, only one B.5.b pump is stored nearby (in the warehouse), while the other is at the fire training facility (farther away). In principle, two B.5.b pumps for two reactors should be sufficient; however, it is not known if there is

[21] There is mention of refilling the FWSTs, but there are no specifics on how this is done.

[22] Treatment of these dependencies were addressed in the selection of MUIEs, in particular. Like most fire SUPRAs, fires in shared spaces and fires that propagate from one unit to another were already identified in the reference plant's fire PRA.

adequate time and other resources to use the second B.5.b pump that is located farther away from the reactors and associated connection points. Consequently, there are questions about the feasibility of both reactors being fed by the B5.b pumps due to potentially inadequate staffing, potentially unavailable equipment to support B.5.b operation, and potentially inadequate time to transport the second B.5.b pump to where it is needed for EDMG strategy implementation.

- For the Phase 3 sitewide dependency assessment of proximity dependencies, three cases were identified as needing consideration:
  - The existing fire PRA has identified scenarios that require both MCRs to be abandoned due to environmental conditions. Specifically, if a fire affects the habitability of the MCR, both MCRs are treated as being affected since they are connected.
  - The existing fire PRA identified scenarios in which a fire can cascade from one unit to the other. When MU risk was calculated, the analyst verified that all credited operator actions were still feasible (i.e., no actions were required in or near the fire location).
  - For Level 2 PRAs, high radiation levels from a reactor post-core-damage are possible in some locations. It is possible that, if such radiation levels existed, that operator actions to implement EDMG strategies for both units could be affected. The operator actions could be delayed (e.g., waiting for health physics personnel to perform radiation surveys) or could be rendered infeasible (i.e., radiation levels too high to attempt performing the action).
- From the Phase 3 sitewide dependency assessment of hazards correlations, operator actions have already been addressed in the existing single unit PRAs for floods, fires, and external hazards. However, if the influence of a hazard encompasses both units, and even in the same way, it is recommended that the operator actions continue to be treated as independent.

4.6.2.2 Implicit Human and Organizational Dependencies

As stated in Sections 4.3.3.3 and 12B.7.3 , potential dependencies other than those described as “explicit” are referred to as “implicit” or “indirect” dependencies. Implicit dependencies could be addressed using modeling assumptions or subjective judgment to adjust human error probabilities (HEPs).

Potential indirect or implicit human and organizational dependencies were identified using the various information collected and interpreted for the existing HRAs performed for the various single unit PRAs. Table 4-8 below documents this evaluation based upon these HRAs, associated plant site visits (including simulator and main control room observations, operator action walk-downs, and operator interviews), and other HRA-relevant information. Modeling these dependencies is beyond the current state-of-practice, but identifying these possibilities is considered good practice. The treatment of potential indirect human and organizational dependencies is a candidate for future research.

As shown in Table 4-8 (and per the discussion in Section 4.3.3.3), there are both positive and negative impacts that are possible for most of these potential dependencies. However, the recommended assessment in EPRI (2021a) is generally that commonalities or dependencies should be considered to have a positive effect.

4.6.3 Results for the Phase 3 Identification of Other Sitewide Dependencies

This section provides a summary of the remaining potential dependencies addressed in the Phase 3 sitewide dependency assessment: proximity dependencies (Section 4.6.3.1), cascading failures (Section 4.6.3.2), and hazards correlations (Section 4.6.3.3). Section 4.6.3.4 identifies some scenarios that required special attention in the development of the MUCDF results. Appendix G provides more detailed results for these potential sitewide dependencies.

Table 4-8 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies

Characteristic of Potential Dependency	Characteristic Exists at Reference Plant?	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Shared MCR	No. The MCRs are connected physically by essentially an “open door,” but they are separated by a relatively large distance with respect to control locations. If Shift Supervisors from each unit wanted to share information, it would only require a short walk.	Because there is considerable separation of control boards and operators for the reference plant, distraction from alarms, etc. from the other unit is very unlikely.	Because travel from the Unit 1 to the Unit 2 MCR is quick and easy, the following is possible: face-to-face communication; “group think” that is correct; sharing “swing” operator; closer coordination between units.	An initial comparison between shared and connected MCRs is documented in EPRI (2021a) and preliminarily did not find any significant differences between the two.
Connected MCR	Yes	Same as above	Same as above	Same as above
Common procedures	Yes. The essentially identical units have essentially identical EOPs, SAMGs, EDMGs, FLEX procedures, fire response procedures, maintenance procedures, etc.	If there was a weakness in the procedures, it likely will affect actions for both units. No such weaknesses were identified in the HRAs performed for the L3PRA project.	Since procedural support for required actions was assessed to be “good,” actions should be independent.	Weaknesses or “gaps” might be considered for explicit modeling (e.g., if an action for one unit is failed due to such a “gap,” then the same action for the second unit probably should also be considered “failed.”
Common training	Yes	Same as for “procedures.”	Same as for “procedures.”	Same as for “procedures.”
Common human-machine interface	Yes			

Table 4-8 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies (cont.)

Characteristic of Potential Dependency	Characteristic Exists at Reference Plant?	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Common command and control (C&C)	Yes and No. Each unit has its own Shift Supervisor. There is one unit supervisor for both units. See "Technical Support Center" for command and control assessment when Emergency Director (ED) responsibilities shift.	Same as "connected MCR"; challenge of responding to multiple reactors within the same time period. However, eventually C&C shifts responsibility to a single ED for both units.	Similar to "connected MCR," common procedures," and "common training."	EPRI (2021a) suggests that on-site command and control should be a net positive.
Common TSC	Yes	By the time the TSC is staffed, the responsibility of ED should be shifted to someone located in the TSC. From 2014 interviews of managers who could take the ED role after transfer into SAMGs,[23] it is expected that the TSC will be staffed with twice as many personnel if the site is responding to a dual-unit event. In addition, the HRA team learned that all four managers who could take the ED role were licensed SROs, or had been licensed SROs, at the reference plant.	Similar to "connected MCR," "common procedures," "common training," and "common C&C." In addition, the HRA team learned during the 2014 plant site visit, that many of those who have responsibilities in the TSC have worked at the reference plant for their whole careers and, therefore, have a strong understanding of the reference plant and its operating history.	Same as for "common C&C."
Common ERO	Yes	Same as for "common C&C" & "common TSC."	Same as for "common C&C" & "common TSC."	Same as for "common C&C" and "common TSC."
Common offsite support	Yes	No information was collected for the L3PRA project on the offsite organization.	No information was collected for the L3PRA project on the offsite organization.	

[23] The 2014 plant site visit for HRA included discussions of potential sitewide events even though the primary purpose was to support Level 2 HRA for internal events PRA.

Table 4-8 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies (cont.)

Characteristic of Potential Dependency	Characteristic Exists at Reference Plant?	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Increased stress due to MU accident	Likely	No specific information regarding stress in MU accidents was collected for the L3PRA project.	No specific information regarding stress in MU accidents was collected for the L3PRA project.	Depending on the severity of the MU event, increased stress could be a reasonable assumption. However, given the advent of FLEX strategies implementation, additional training and attention may offset potential stress for some severe accidents.

4.6.3.1 Proximity Dependencies

At this time, with only multi-unit Level 1 risk results developed, no specific scenarios have been identified that definitively involve proximity dependencies alone. However, the search for proximity dependencies was revisited during the development of multi-unit Level 2 risk results (see Section 8) and when multi-source scenarios were developed for both reactors and the SFPs (see Section 9). Also, the section below on hazards correlations overlaps the assessment of proximity dependencies.

The following contexts that result in SSC failures were searched for and are used again in later stages of the ISR task:

- common conditions for SSCs for both reactors (e.g., due to the same hazard or response to the same hazard)
- conditions created by one reactor that affects SSCs for the second reactor

Commonality for the two reactors on the reference site include:

- identical or similar design (e.g., layout and design of the plants, dimensions or sizes of SSCs)
- common or shared locations (such as those identified in the Phase 2 sitewide dependency assessments)
- traditional application of hazard correlations (e.g., modeling identical response for both reactors to the same external hazard)

However, the likelihood of proximity dependencies for the reference site is limited by:

- separation or independence of most SSCs modeled (i.e., the Phase 2 sitewide dependency assessment indicated that there are few shared or connected SSCs between the two reactors on the reference site)
- few conditions (i.e., only those caused by fires, internal floods, external hazards, or radiation) can catastrophically affect SSCs in both reactors

A consequence of the above limitations is that proximity dependencies for SSCs due to environmental conditions (that are not associated with external hazards) can only occur for the reactors at the reference site if SSCs for both reactors are shared or connected.

4.6.3.2 Cascading Failures

As for proximity dependencies, the focus of the Phase 3 sitewide dependency assessment for cascading failures was on the two reactors. The SFPs do not share any support systems with the reactors and other potential cascading failures (e.g., fires, internal flooding events) are not important to the SFPs. Similarly, because the DCS facility is independent of other radiological sources and is remotely located, it is unlikely that failures could cascade from the DCS to the other radiological sources. However, this type of dependency was revisited for the MU Level 2

PRA work (see Section 7) and future work could address potential failures that could cascade from the SFPs to the reactors.[24]

Except for certain fire scenarios that were identified in Table 4-3 as multi-unit initiating events (see Appendix C for further discussion), failures of one unit propagating to another unit are not expected for the reference site. This expectation is based primarily on the determination in the Phase 2 sitewide dependency assessment that the two reactors are only loosely coupled.

4.6.3.3 Hazards Correlations

Potential dependencies between the two reactors with respect to hazards correlations (and/or proximity dependencies) for external hazards were assessed. Such dependencies related to external events and the SFPs are addressed in the development of sitewide scenarios in a later ISR task, as are dependencies associated with conditions associated with Level 2 PRA.

In summary, the following were identified in this Phase 3 sitewide dependency assessment as requiring attention for the ISR task:

- Seismic correlation for SSCs between two units should be considered whenever necessary for multi-unit seismic initiating events. (Potential inter-unit seismic correlation is related to both the hazard and the proximity, as noted above.) As the intensity of the seismic event pga increases (e.g., for higher seismic bins), the likelihood of MU seismic correlation increases. Although a seismic correlation model exists for the single unit SSCs and is already included, a two-unit seismic correlation model does not exist. A simple two-unit seismic correlation model has been developed using the Unit 1 CDF cutsets.

- During the wind-events walkdown, no major safety system failures due to wind events (including those that could affect both units simultaneously) were identified. However, the walkdown scope did not include examination of the impact of a wind-related structure failure on another structure belonging to the other unit. Currently, no wind-related multi-unit failures due to proximity are envisioned. (However, there is a scenario involving switchyards listed below for potential scenarios that require “special attention.”)

- External flooding was assessed in the L3PRA project for Unit 1 among the events collectively gathered under “other hazards” (NRC, 2023a), without detailed modeling. Other than possible impact on the turbine building shared by both units (proximity), this hazard is not further pursued for multi-unit impact (hazard- and proximity-wise) due to its expected lower risk as compared to other multi-unit events. Also, the NRC, in a safety evaluation, accepted the reference plant’s flooding focused evaluation conclusion that external flooding could be screened out.

- There are other hazard categories included to some degree of detail in the L3PRA documentation (NRC, 2023a). These other categories are deemed to cause lesser risk (than those categories modeled in detail) and are not evaluated here. However, it is

[24] During development of the SFP Level 1 and 2 PRAs, scenarios that involve implementation of procedures that use water inventory from the reactors to restore water level in the SFPs were discussed.

recognized that airplane accidents have the potential to simultaneously damage multiple structures belonging to two units.

- Development of MU Level 2 risk results (see Section 7) also involved consideration of hazards correlations.

4.6.3.4 Scenarios Requiring Specific Attention

The following scenarios received special attention in the development of the MUCDF results:

- MCR failures with or without MCR abandonment—Since the MCRs are connected, an internal fire event impacting the MCR for one unit has a strong potential to impact the other MCR, both HRA-wise and equipment-wise. If an MCR evacuation scenario for one unit occurs, a complete correlation between the two units can be assumed. This is an example of a hazard and proximity-related case.

- Potential two-unit interactions during SBO events—If both units are in SBO, various local actions (e.g., actions away from the MCR) are expected to be ongoing during the same time windows. It is possible that these actions may impact each other. For example, if ELAP is declared in both units, even if the FLEX building housing the equipment for both units is not damaged, it would be a single point of focus for both crews for access and for moving equipment. This could affect crew performance, though it would be challenging to quantify the actual impact.

- Failures affecting the common low voltage switchyard (and the high voltage switchyard)—Since both units share the switchyards, equipment failures due to hazards (like seismic events, wind-events, external flooding, LOOPWR, even switchyard fires) may be impacted both by the hazard and proximity. In most cases, the outcomes are expected to impact LOOP initiating event frequencies and AC recovery probabilities.

4.7 Summary of Sitewide Dependency Assessment Results

From the results given in the previous sections:

- The two reactors on the reference site are mostly independent (i.e., “loosely coupled”), except for some identical components and shared hazards for external events.

- There is some resource sharing between the SFPs and the reactors for seismic events.

- There are no dependencies identified between the DCS facility and the two reactors and the SFPs.

- The sitewide dependencies that have been identified provide important insights by themselves. In addition, these results are used to develop multi-unit risk results and, later, multi-source risk results.

5 MULTI-UNIT AND SITEWIDE INITIATING EVENTS ANALYSIS

This section provides the MUIEs and sitewide IEs that have been selected for the ISR task and their associated frequencies. Sections 12C.3 and 12C.4 provide more details. This is the third step in the overall ISR task.

5.1 Selection of MUIEs and Sitewide IEs

Several factors were considered for the ISR task in selecting which MUIEs and/or sitewide IEs are relevant to MU and multi-source calculations. In particular, the number of radiological sources affected by the initiator, as well as the initiator's percentage contribution to the risk of the individual radiological sources, were important in this selection process. Resource constraints for the overall L3PRA project were another important factor.

The principal basis for the selection of MUIEs and sitewide IEs was the results of the Phase 1 sitewide dependency assessment documented in Section 4.4. In particular, only those MUIEs and/or sitewide IEs that can impact two reactors (as well as those that can impact either the SFPs or DCS) were selected. Note that, from the Phase 1 sitewide dependency assessment, the only relevant initiators for the SFPs are seismic events. Consequently, when multi-source scenarios are developed and multi-source risk estimated, the SFPs only contribute to results associated with seismic events.

Table 5-1 below shows the IEs that have MU and/or sitewide impact and that were addressed by the ISR task.

5.2 Calculation of MUIE and Sitewide IE Frequencies

There are differences between single unit IE frequencies (IEFs) in how they were calculated, and the data used in those calculations. Some of the MUIEFs or sitewide IEFs not only impact the entire reference site, but also were initially developed as sitewide frequencies. All IE frequencies for external hazards (e.g., seismic events) were developed in this way. However, per PRA convention, even these IEFs were reported in units of "per-critical-year. [25] Consequently, the originally determined frequency for these IEs was used directly in MU risk calculations. In addition, the original IEF was used for certain fire scenarios (e.g., main control room abandonment scenarios and fires that cascade from one unit to another).

Other MUIEs or sitewide IEs frequencies needed to be adjusted for MU risk calculations. The IEs that needed adjustments are LOOPs and loss of nuclear component service water.

[25] Typically, a capacity factor is used with IE frequencies that have been developed in this way. The L3PRA project did not use capacity factors in its PRAs. However, since the capacity factor for the reference plant is high (i.e., 0.93), the difference between reactor-critical-year and reactor-calendar-year is well within uncertainty bounds.

Table 5-1 List of IEs That Have Potential Multi-Unit or Sitewide Impacts

No.	Scenario Name	Scenario Description	MU Scenario Characteristics
1	MU-IE-LOOPGR	Grid-Related LOOP	SBO and AC power recovery failure
2	MU-IE-LOOPPC	Plant-Centered LOOP	"
3	MU-IE-LOOPSC	Switchyard-Centered LOOP	"
4	MU-IE-LOOPWR	Weather-Related LOOP	"
5	MU-LONSCW	Loss of NSCW	Loss of NSCW in both units
6	MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned with CCDP =1
7	MU-IE-FRI-2	Shared (A+Y) area fires by Unit 1 and Unit 2	at least MU LOOP
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least MU LOOP
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least MU LOOP
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic BIN-1
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic BIN-2
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic BIN-3
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic BIN-4
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic BIN-5
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic BIN-6
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2-unit SBO and major structural damage (EQK-BIN7) with CCDP =1
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and major structural damage (EQK-BIN8) with CCDP = 1
18	MU-IE-WIND-1	SBO and SSC wind damage	SBO and wind damage to SSCs

Regarding LOOPs, there are four types of LOOPs to be addressed, each of which were originally developed to apply to single units (even if there were multiple units on a site):

- grid-related LOOPs (LOOPGRs)
- plant-centered LOOPs (LOOPPCs)
- switchyard-centered LOOPs (LOOPSCs)
- weather-related LOOPs (LOOPWRs)

A variety of approaches have been used or proposed for developing MU or sitewide IE frequencies for LOOPs. Examples of such approaches are given in IAEA (2019)[26] and EPRI (2021a).

IAEA (2019) provides three different approaches for calculation of LOOP MUIE frequencies (MUIEFs). It identifies various limitations in these approaches, including inability to account for plant variability and lack of relevant data. IAEA (2019) also recommends separation of LOOP

[26] The more recent IAEA report on MUPRA (IAEA, 2021a) references IAEA (2019) in its discussion of calculating MUIEFs.

event data into categories of reactor-centered, site-centered, and region-centered categories for calculation of MUIEFs.

The EPRI report on MUPRA (EPRI, 2021a) describes the purpose of the data analysis task for its approach as the identification of types of MUIEs that appear in operating experience and the estimation of the likelihood of MU events relative to SU events on MU sites. However, the EPRI report cautions that “[t]his data is not used to estimate actual MU initiator frequencies; that is a site-specific task.” Instead, the EPRI report describes calculations for MU LOOP frequencies on a “per unit” and “per site” basis, using international reactor trip data over a 10-year period and adjusting for times when the reactor was not operating. The EPRI report (2021) describes the MUIEF results for this data-driven approach and how to derive a conditional probability of a MU trip, given a SU trip, using the same data.[27] However, in its conclusions regarding data analysis, the EPRI report recommends using generic fractions (which appear to function as conditional probabilities) to develop MUIEFs from SUIE frequencies due to limited relevant data.

The approach used for the L3PRA project’s ISR task is similar to that used in the EPRI report (2021a). In particular, the MU conditional probabilities used in the ISR task are taken from Table 17 of the 2021 version of Idaho National Laboratory’s (INL’s) “Analysis of Loss-of-Offsite-Power Events Update” report (INL, 2007). However, unlike the EPRI report’s use of international data, INL used only U.S. data (2006 through 2020) to develop MU conditional probabilities. The ISR task uses the mean values shown in the INL report.[28] Note that the INL data analysis, unlike the EPRI report’s analysis, indicates that even plant-centered LOOPs can result in a MU event.

For LOOPs, the MUIEF is calculated through use of a multiplier. A Unit 2 (U2) multiplier is introduced to calculate a two-unit scenario initiating event frequency. This multiplier was multiplied by the Unit 1 IE frequency (U1-IEF) to obtain a MUIEF. The multiplier is 1.0 if the Unit 1 (U1) IE causes also a U2 trip. If a fraction of the U1 IEs causes a U2 trip, the multiplier is equal to the fraction. The multiplier cannot be greater than 1.0.

Although not explicit in the INL report (INL, 2007), the multipliers were used to adjust SUIE frequencies to MUIEFs were interpreted to be in units of “per-reactor-critical-year.”[29]

For the loss of nuclear component service water, the L3PRA project has used an IE frequency based on a common cause failure (CCF) analysis. For the MUCDF results developed at this time, complete dependency was assumed between the NSCW pumps such that the SUIE frequency is used as the MUIEF, too (i.e., a multiplier of 1.0).

[27] The EPRI data analysis for developing MUIE frequencies is shown to have units of “per-site-year” using international data consistent with “per-reactor-operating-year” units.”

[28] Note that the ISR task uses MU conditional probabilities based on data in 2021 updated report. However, the L3PRA project’s PRA models have a freeze date of 2012 so they use an earlier version of LOOP data for the single unit IE frequencies.

[29] Table 17 in the 2021 update (INL, 2021) to the original INL data analysis report (INL, 2007) shows that the only trip data used to develop MU conditional probabilities is for operating (i.e., not shutdown) reactors.

5.3 Overall Results for MUIE and Sitewide IEs and Their Frequencies

Table 5-2 provides the final MUIE frequencies. The table also includes the single unit IE frequencies, along with the multiplier (if applicable) used to develop the corresponding MUIEF. The following points relate to the information provided in Table 5-2:

- The units for all IE frequencies (both single unit and multi-unit) are in terms of "per reactor-critical-year."
- The single unit IE frequency for loss of NSCW was used as the MUIEF (i.e., MU multiplier of 1.0), based on the conservative assumption that if all six NSCW pumps in one unit fail due to a common cause, the six pumps in the other unit will fail from the same common cause.
- The MUIEFs can also be considered as sitewide IE frequencies, where applicable (e.g., the seismic events contribute appreciably to both reactor and SFP risk).

Table 5-2 MU and Sitewide Initiating Event Frequencies

	Scenario Name	U1IEF (/rcy)	MU Multiplier	MUIEF (/rcy)
1	MU-IE-LOOPGR	1.23E-02	0.500	6.15E-03
2	MU-IE-LOOPPC	1.93E-03	0.056	1.07E-04
3	MU-IE-LOOPSC	1.04E-02	0.269	2.80E-03
4	MU-IE-LOOPWR	3.91E-03	0.625	2.44E-03
5	MU-LONSCW	3.47E-05	1	3.47E-05
6	MU-IE-FRI-1	1.50E-07	1	1.50E-07
7	MU-IE-FRI-2	3.40E-02	1	3.40E-02
8	MU-IE-FRI-3	9.10E-03	1	9.10E-03
9	MU-IE-FRI-4	9.10E-03	1	9.10E-03
10	MU-IE-EQK-1	1.60E-03	1	1.60E-03
11	MU-IE-EQK-2	2.20E-04	1	2.20E-04
12	MU-IE-EQK-3	4.80E-05	1	4.80E-05
13	MU-IE-EQK-4	1.30E-05	1	1.30E-05
14	MU-IE-EQK-5	4.30E-06	1	4.30E-06
15	MU-IE-EQK-6	1.90E-06	1	1.90E-06
16	MU-IE-EQK-7	2.50E-07	1	2.50E-07
17	MU-IE-EQK-8	2.30E-09	1	2.30E-09
18	MU-IE-WIND-1	8.89E-03	1	8.89E-03
	TOTAL			**7.45E-02**

*rcy – reactor-critical-year

6 MULTI-UNIT LEVEL 1 CORE DAMAGE FREQUENCY ESTIMATES

This section describes the fourth step in the overall ISR task. Section 6.1 describes a PRA-software-based approach for obtaining multi-unit Level 1 PRA core damage frequency (MUCDF), based on development of a traditional event tree-fault tree multi-unit (MU) PRA model, as well as a variation of this approach that uses the cutsets obtained from the single unit (SU) PRAs (SUPRAs). Since the L3PRA project team was unable to implement either approach due to limitations in the NRC's PRA software tool at the time of the analysis, Section 6.2 describes a simplified approach that was implemented for the L3PRA project. Section 6.3 provides a summary of the MUCDF results obtained using this simplified approach.

Appendices H, I, and J provide supporting details for the development of MUCDF results for the L3PRA project. Appendix H provides background on the coupling factors that were used to represent various sitewide dependencies. Appendix I provides further details on the MUCDF calculation approach and its results. Appendix J documents alternate calculations performed to test the MUCDF calculation approach used for the ISR task.

6.1 PRA-Software-Based Approach for MUCDF Estimation

Step 2 of the overall ISR task identifies dependencies between the reactors, as well as between the reactors and other major radiological sources on the site. Note, if there are no dependencies between the reactors on the site, calculation of MUCDF would be trivial and limited to identifying what initiating events (IEs) trip all reactors on the site. However, the sitewide dependency assessment will assuredly identify additional important types of cross-unit dependencies for the reactors (i.e., beyond multi-unit initiating events [MUIEs]). Examples of potential MU dependencies include: (1) cross- or inter-unit CCFs, (2) certain basic events (BEs) associated with the recovery of offsite power, (3) human failure events (HFEs), and (4) correlated seismic failures across units.

Given that MU dependencies are likely to exist, the specific high-level steps for calculating MUCDF using a traditional PRA logic model and PRA software include:

1. identify MUIEs
2. determine the MUIE frequencies (MUIEFs)
3. develop an MU event tree for each identified MUIE
4. identify MU dependencies to be addressed
5. determine the appropriate values to assign to the BEs that represent MU dependencies
6. calculate MUCDF for each MUIE

Steps 1 and 2 were addressed in Sections 4 and 5.

Step 3 can be accomplished for each MUIE by linking the sequences for the Unit 1 event tree to the corresponding Unit 2 event tree. The end-states of the Unit 2 event tree can be designated as no core damage ("OK"), Unit 1 core damage, Unit 2 core damage, or multi-unit core damage. This linked two-unit PRA model needs to account for dependencies between the two units that were identified as part of the sitewide dependency assessment. Step 4 is accomplished using the information obtained from the sitewide dependency assessment (described in Section 4) and review of the PRA model. Step 5 is an area of significant uncertainty due to data limitations, particularly with respect to data on large CCF groups. This step will likely require a substantial amount of analyst judgment (possibly involving the use of expert elicitation). There are undoubtedly multiple ways to accomplish Step 6. One possibility is to use PRA software to quantify the two-unit linked event tree model and then use cutset post-processing rules to substitute MU versions of the SU basic events. While the identification of the necessary substitution rules would require manual review of the SU (or MU) cutset listings and analyst judgment, the actual substitution and requantification process would be automated through use of the PRA software. And, as with the quantification of all PRA models, the analyst will need to manually review the final cutset listing, in this case as a further check that all significant MU dependencies have been addressed and that the substitution rules have been applied correctly.

If full quantification of the two-unit linked event tree model is not practical, other approaches for estimating MUCDF could be implemented. One possible alternative approach, which also makes use of PRA software, is to obtain the MU cutsets through manipulation of the SU cutsets. With this approach, Steps 1, 2, 4, and 5 are retained from the previous approach, but Step 3 is replaced by the following:

- A large fault tree is created for each SUPRA (e.g., Unit 1 and Unit 2 PRA) model, where the fault tree consists of a large OR-gate with the individual cutsets as inputs to the OR-gate.

- Each cutset is, in turn, an AND-gate, where the individual cutset BEs are the inputs to the AND-gate.

- A fault tree is created for the MUPRA model, where the top event is an AND-gate, and the two SUPRA fault trees are the inputs to the AND-gate.

- The MUPRA fault tree is solved to obtain the MUPRA cutsets.

Step 6 from the previous approach is also retained, but the MUPRA cutsets are obtained from solving the MUPRA fault tree, as opposed to solving the two-unit linked event tree model. Similar to the previous approach, cutset post-processing rules are used to substitute MU versions of the SU basic events to account for the MU dependencies (also with this approach, post-processing is needed to replace the two SU initiating event frequencies [SUIEFs] with the corresponding MUIEFs). And, as with the previous approach, the analyst will need to manually review the final cutset listing as a further check that all significant MU dependencies have been addressed and that the substitution rules have been applied correctly.

Discussion of similar approaches for MUCDF estimation can be found in EPRI (2021a) and IAEA (2019, 2021).

6.2 Simplified Approach for MUCDF Estimation

The L3PRA project team investigated various viable and acceptable approaches for calculating MUCDF for the two, essentially identical reactors on the reference site. For example, project team members were involved in international workshops and reports (e.g., IAEA 2019) related to MU risk, and have reviewed both domestic (e.g., EPRI, 2021a) and international efforts (e.g., IAEA, 2021) related to MU risk. In addition, various trial calculations and proofs of concept have been performed as part of the L3PRA project. These trials included exploration of using NRC's PRA software tool, SAPHIRE, to build a traditional event tree-fault tree MUPRA model. However, at present, the SAPHIRE software is limited in its ability to produce MU risk results for a full-scale MUPRA model.[30]

The approach ultimately selected for calculating MUCDF in the L3PRA project has been labeled the "cutset estimation method" (CEM). Development of this approach was based on a thorough understanding of the plant-specific, potential dependencies between the two reactors on the reference site, the cutsets from the SUPRAs, and the potential impact on MU risk calculations from coupling factors between the two reactors. Trial applications of the CEM approach led to iterations to the approach. The sections below summarize the plant-specific factors and other considerations that supported development of this simplified approach for estimating MUCDF.

6.2.1 Motivation for Approach

There are several conditions that support use of a simplified approach, such as the CEM approach, for calculating MUCDF for the L3PRA project. These conditions include:

- At-power, SU cutset results are already available for Level 1 PRAs for all hazards.
- Since the two units on the reference site are essentially identical, there is no need to develop a separate Unit 2 PRA model. In addition, differences between the units were not relevant for the selected MUIEs and associated MUPRAs. Consequently, Unit 1 PRAs and associated results can be used to represent Unit 2.
- There are state-of-the-art limitations for the assignment of MU coupling factors (e.g., hazard correlations) and the identification and assessment of uncertainties in MUCDF calculations, which tends to support the use of simpler, more cost-effective methods.
- The NRC's PRA software tool, SAPHIRE (INL, 2011), is limited in its capabilities for large models (e.g., a complete, two-unit, event tree-fault tree PRA model).

In addition, there are several factors that support the use of the CEM specifically for the two reactors on the reference site, especially:

- The results of a sitewide dependency assessment show that there are few dependencies between the two units on the reference site and the principal dependencies that do exist for most of the MUIEs addressed are generally limited to those associated with a common location or similar design (e.g., shared switchyard, common initiating events, common response to external events, and potential cross-unit CCFs). (Most of the seismic bins are an exception to this generality.)

[30] Subsequent to the estimation of MUCDF, SAPHIRE was modified to be able to perform the alternative (cutset manipulation) approach described at the end of Section 6.1. While this SAPHIRE modification was not implemented in time to allow its use for estimating MUCDF, it was used for estimating MU release category frequencies as part of the MU Level 2 PRA (see Section 8.3).

- In most cases, only a few hundred cutsets are needed to represent 95 percent or more of the single unit CDF (SUCDF) results for the full range of hazards.

6.2.2 The Importance of Multi-Unit Dependencies

As presented earlier in Section 4, a formal, systematic sitewide dependency assessment was performed for all major radiological sources on the reference site (i.e., two reactors, two spent fuel pools, and the dry cask storage facility). This dependency assessment was performed for all hazards, and for both Level 1 and Level 2 PRAs, and revealed that there are very few dependencies between the two reactor units (as well as between the reactors and the other major radiological sources). As stated in Section 6.1, the dependencies between the two units are mostly those that cannot be avoided (e.g., common IEs such as losses of offsite power [LOOPs] and external hazards, a common switchyard, and potential for cross-unit CCFs). As a result, only a limited number of cross-unit dependencies need to be represented in the MUCDF calculations. In addition, cutset reviews performed as part of the calculation of MUCDF identified specific scenarios and BEs that needed to be addressed. The cross-unit dependencies relevant to the calculation of MUCDF fall into the following sitewide dependency categories:

- initiating events that cause both reactors to experience reactor trip (i.e., MUIEs), such as:
    - all four types of LOOP
    - loss of nuclear service water cooling water (NSCW)
    - external hazards (e.g., high wind and seismic events)
- cross-unit CCFs such as:
    - group expansion of existing (i.e., SU) CCFs
    - new CCFs (e.g., CCF of the turbine-driven auxiliary feedwater [TDAFW] pumps at both units [each unit has one TDAFW pump])
- a limited number of BEs associated with offsite power restoration (for grid-related and weather-related LOOPs only)
- a limited number of HFEs related to, for example:
    - starting NSCW pumps or tripping reactor coolant pumps (RCPs) for loss of NSCW events
    - establishing high pressure recirculation and initiating cooldown following a seismically-induced small-break loss of coolant (SLOCA) accident (for bin 1 seismic events only)
- hazard correlations (e.g., seismic correlations)

Each of these types of MU dependencies needs to be represented in the MUCDF calculations. It should be noted that, based on the sitewide dependency assessment, the principal cross-unit dependency of concern for most MUIEs is expected to be cross-unit or MU CCFs. However, hazard correlations are expected to dominate the results for most of the seismic bins modeled in the L3PRA project.

For the L3PRA project, the results of the sitewide dependency assessment were essential in supporting the decision to use a simplified approach (i.e., the CEM approach) for calculating MUCDF. EPRI (2021a) also states that it is appropriate to use simplified approaches for calculating MUCDF if the reactors are "loosely coupled" (i.e., have very few dependencies).

6.2.3 Use of SU Cutsets to Estimate MUCDF

The CEM approach for estimating MUCDF was developed for the L3PRA project to address specific aspects of the reactors on the reference site. This approach was used in place of a PRA-software-based approach due to limitations in the SAPHIRE PRA software and to limit the level of effort.

In applying the CEM approach for the L3PRA project, MUIEs due to internal events were addressed first, specifically LOOPs and losses of NSCW. The four identified MUIE fire scenarios were already addressed, either in full or in part, for the SUPRA, so little additional modeling was needed for these scenarios. The calculations of MUCDF for seismic events built on those for internal events, but ultimately was dominated by extension of the SUPRA seismic hazard correlations to the MUPRA. The treatment of wind events was generally the same as for LOOPs, except that more SU cutsets were needed to represent 95 percent of the SUCDF for all wind events.

Note, if there were no dependencies between the two reactors on the reference site, calculation of MUCDF would be trivial and limited to identifying what initiating events trip both reactors. However, for internal events, the sitewide dependency assessment identified three additional important types of cross-unit dependencies for the reactors on the reference site: (1) cross- or inter-unit CCFs, (2) certain BEs associated with the recovery of offsite power, and (3) several HFEs.

As mentioned in Section 6.1, when using a two-unit PRA logic model, these MU dependencies would likely be addressed through the use of cutset post-processing rules to substitute MU versions of the SU basic events. While the identification of the necessary substitution rules would require manual review of cutset listings and analyst judgment, the actual substitution and requantification process would be automated through use of the PRA software. The CEM also requires manual review of cutset listings and analyst judgment to identify the cutsets that have MU dependencies and determine how they should be addressed. However, in lieu of the substitution rules used for the PRA software, the CEM addresses MU dependencies through the manual application of coupling factors. The CEM uses Excel spreadsheets to list the SUPRA cutsets, incorporate the coupling factors, and quantify the MUPRA cutsets.

Additional considerations for the application of the CEM for the L3PRA project include:

- The primary focus for LOOPs (and loss of NSCW and wind events) is on MU CCFs, so SUPRA (i.e., Unit 1 PRA) cutsets containing CCFs were flagged for specific calculational adjustments.

- Since the Unit 1 PRA results (e.g., CDFs for individual cutsets containing CCFs and overall conditional core damage probability [CCDP]) are used by the CEM, the coupling factors used are conditional probabilities (e.g., the conditional probability of a Unit 2 CCF, given that an identical Unit 1 CCF has occurred).

- For each Unit 1 PRA cutset that contains a CCF basic event, the following two contributions to MUCDF were calculated (and the rare events approximation was used for the Boolean addition of these contributions):

    - MU CCF of identical components in both units (e.g., an MU CCF involving all four emergency diesel generators [EDGs] failing to start; in other words, both EDGs in each unit fail to start), capturing the cross-unit dependent contribution

    - the SUPRA cutset involving the single unit CCF multiplied by the Unit 2 CCDP for the specific IE being analyzed (i.e., the SUPRA CDF for the specific IE being analyzed divided by the SUIE frequency [SUIEF]), capturing the cross-unit non-dependent contribution

Based on the above, a limitation of the CEM is that standard BE importance measures (e.g., Fussell-Vesely importance or risk achievement worth) cannot be calculated for the MU model. Since conditional probabilities are used to represent the response of the second unit to the MUIE (as opposed to the actual cutsets for the second unit), a complete set of MUCDF cutsets is not available to perform traditional importance analysis.

The specific high-level steps for applying the CEM for calculating MUCDFs are as follows:

1. identify MUIEs

2. determine the MUIE frequencies (MUIEFs)

3. identify cutsets that make up 95 percent of the SUCDF for each MUIE

4. review cutsets and identify the different BEs that need to be addressed to account for MU dependencies

5. assign cross-unit BE coupling factors (or hazard correlations) for each relevant cutset

6. calculate associated cutset coupling probabilities (as needed)

7. calculate MUCDFs

Steps 1 and 2 were addressed in Sections 4 and 5 and are performed in the same way as for the PRA-software-based approaches described in Section 6.1. Steps 3 through 7 are unique to the CEM approach although some are similar to steps performed using the PRA-software-based approaches (particularly, Steps 4 and 6). Steps 3 through 7 are discussed further below.

6.2.3.1 Identify SU Cutsets to Represent SUCDF in MUCDF Calculations (Step 3)

After identifying the MUIEs to address, the next step is to select the cutsets that represent the great majority of the SUCDF. Ideally, this set of cutsets represents approximately 95 percent of the SUCDF for the MUIE in question. The goal is to identify a manageable number of cutsets to use in MUCDF calculations while still representing most of the SUCDF.

In order to perform CEM calculations in later steps, the identified cutsets were exported to an Excel worksheet. Cutset information that was exported to the Excel spreadsheet included:

- cutset number
- cutset CDF
- percentage of total CDF contribution for each cutset
- elements of the cutset:
    - IE identifier, description, and frequency
    - BE identifiers, descriptions, and probabilities

In Step 7, a scale-up factor is developed based on the percentage of SUCDF represented by the selected SU cutsets. This scale-up factor is used to adjust the MUCDF results from that based on 95 percent of the SUCDF results to a result that approximates 100 percent of the SUCDF results.

6.2.3.2 Review SU Cutsets and Identify BEs to Modify to Represent MU Dependencies (Step 4)

The review of the SUPRA (i.e., Unit 1) cutsets is an important step in developing and implementing CEM for MUCDF calculations. The review may involve multiple iterations, with each iteration providing further information for developing and applying coupling factors. The decision on how many iterations to perform is based on weighing the anticipated improvement in the accuracy of the MUCDF estimation versus the level of effort involved. This decision necessarily relies heavily on analyst judgment.

The main objective of cutset review is to identify those cutsets that involve potential MU dependencies (based on the results of the sitewide dependency assessment). The cutset reviews also provide the analyst with insights into which SU cutsets are important contributors to MUCDF. This is important to limit the level of effort in identifying and applying coupling factors.

Cutset reviews are also important for identifying cutsets that include more than one BE that has a potential MU dependency. For the L3PRA project, most of these cutsets were for seismic events. As discussed further in Section 6.2.3.4, the treatment of MU dependencies for cutsets with multiple dependent BEs requires the development of coupling factors on a cutset basis (as opposed to just on a BE basis).

Documentation of the cutset review results in Excel spreadsheets supports later CEM steps and facilitates independent, technical review of the analyst’s implementation of the CEM approach. Note, as mentioned earlier, if a PRA-software-based approach is used instead of the CEM approach, cutset reviews would still be necessary to identify the necessary post-processing rules for addressing cross-unit CCFs and other types of MU dependencies, though the actual substitution and requantification process would be automated through use of the PRA software.

For application of the CEM approach for LOOPs, the loss of NSCW, and wind events, the important MU dependencies are MU CCFs and, therefore, they are the focus of the initial cutset review. MUCDF results were produced for seismic events in a similar way except with focus on hazard correlations assigned for the SUPRA and the potential for similar impact on both units simultaneously.

For LOOPs (as well as loss of NSCW and wind events), all CCFs already modeled in the SUPRA and identified in the Phase 3 sitewide dependency assessment (see Section 4.6.1.1) were considered for modeling as MU CCFs. Also, as identified in the sitewide dependency assessment, there is one component type that needs to be represented in MUCDF estimates that was not part of an existing CCF group. Namely, each unit on the reference site has one TDAFW pump. Consequently, SU cutsets containing TDAFW pump failures were tagged for representation as MU CCFs.

Each occurrence of a CCF for a particular component and failure mode was noted. The "search" feature in Excel was used to identify existing SU CCFs, as well as TDAFW pump failures (both "fails to run" and "fails to start"). The Excel spreadsheets were edited to highlight the SU CCFs and TDAFW pump failures that would be later treated as MU CCFs. This highlighting assisted not only the cutset review process and later MU CDF calculations, but also facilitated internal technical reviews of how the CEM approach was implemented.

The Excel spreadsheet search feature also was used to identify other pertinent information, for example:

- the total number of occurrences for each CCF for each relevant system, component type, and failure mode in the SU cutsets

- the total number of occurrences of TDAFW pump fails to run

- the total number of occurrences of TDAFW pump fails to start

Each cutset was then labeled in a new Excel spreadsheet column with a "Cutset Type." For example, each cutset that contained a CCF was labeled either "CCF1" or "CCF2," consistent with the CCF types and associated coupling factors presented in Section 6.2.3.3 and Table 6-1 below.

For the initial cutset review, all cutsets that did not contain a potential MU CCF event were assigned cutset type "RANDOM."[31] For the MUIEs evaluated in the L3PRA project, cutsets that contain MU dependencies and do not contain at least one CCF event are very rare and do not make a significant contribution to MUCDF.

If the analyst judges the initial cutset review to be sufficient to obtain a reasonable estimate of MUCDF, then the next step taken by the analyst is the assignment of BE coupling factors (described in Section 6.2.3.3). However, if a more accurate estimation of MUCDF is desired, additional cutset review can be performed. In particular, the SU cutsets that are already flagged as having a potential MU CCF can be further analyzed to see if additional BEs in the cutsets should be accounted for. These other BEs fall into one of the following categories:

- independent or "random" BE, such as:
    - random equipment failures

[31] It should be noted that not all cutsets in the Excel spreadsheets are labeled (or treated) consistent with the guidance in this section. To limit the level of effort to implement this proof-of-concept study, analyst judgment was used to determine how completely to address the cutsets. As demonstrated by a sensitivity analysis for the LOOPWR MUIE, a more rigorous implementation of the guidance (i.e., for all cutsets in the top 95 percent of SUCDF) would not change the estimated MUCDF appreciably.

    - independent HFEs

- dependent BEs, such as:

    - dependent HFEs
    - BEs associated with the recovery of offsite power that are common for both reactors

Analyst judgment is used to determine how many, and which, additional SU cutsets to address. For these multi-element cutsets, a “cutset coupling probability” is calculated (see Section 6.2.3.4), reducing the conservatism of results caused by considering only cross-unit CCF coupling factors.

For the L3PRA project, the sitewide dependency assessment identified BEs associated with restoration of offsite power that were considered common to both Units 1 and 2 and assumed to be fully dependent. In addition, an understanding of the two plants at the reference site and their operation was used to identify which HFEs were fully dependent, partially dependent, or independent. Cutsets containing operator actions common to both units were labeled with “HEP.” Also, some cutsets that contained both a CCF and an independent operator action were labeled as cutset type “CCF1-HEP” (or “CCF2-HEP”) and some cutsets that contained combinations of CCFs and random events were assigned a cutset type label of “CCF1-RAND” or “CCF2-RAND.” Again, analyst judgment was used to determine which cutsets to label and address. Section 6.2.3.3 provides further discussion of the assignment of cutset types and their use in the CEM approach.

It is important to note that the quantification results provided in this report for the proof-of-concept analyses generally only considered the initial iteration of cutset review. However, a sensitivity analysis was performed using LOOPWR to demonstrate that it would not appreciably reduce the estimated MUCDF if additional iterations were performed.[32] It also should be noted that less rigor was used for lower-contributing MUIEs (e.g., LOOPPC) to limit the level of effort.

6.2.3.3 Assign Cross-Unit Basic Event Coupling Factors for Each Relevant SU Cutsets (Step 5)

As discussed above, the ISR task identified several types of MU dependencies for the two reactors on the reference site that should be represented in MUCDF calculations. Cross-unit coupling factors were used in the ISR task to represent three of those types: cross-unit (or MU) CCFs and certain human dependencies (Section 6.2.3.3.1) and cross-unit seismic hazard correlations (Section 6.2.3.3.2).[33]

6.2.3.3.1 Assignment of Cross-Unit Coupling Factors for MU CCFs and Human Dependencies

For most of the MUIEs addressed in the ISR task, the focus of the CEM approach for estimating MUCDF is on SU cutsets that contain CCFs. In traditional SUPRAs, CCFs are modeled such that cutsets are generated that include both the simultaneous random failures of two (or more), identical components and the CCF of those components. An analogous approach is used for

[32] Using the CEM, LOOPWR MUCDF was calculated to be 4.47E-7/rcy. In the sensitivity analysis involving additional iterations of cutset review, the LOOPWR MUCDF was calculated to be 4.35E-7/rcy.

[33] IAEA (2021a) discusses seismic hazards correlations as representing a type of “common cause” for component or structural failures.

MUPRA, where two types of contributions have to be accounted for: (1) simultaneous dependent failures between both units (i.e., cross-unit CCFs) and (2) failures that are not dependent between both units (including both random failures and CCFs that do not propagate to the other unit).

When using traditional PRA software, it may be possible (depending on the size of the PRA models and the capabilities of the software) to combine the Unit 1 and Unit 2 SUPRAs, which will automatically account for the various cross-combinations of independent and dependent failures. As discussed later in Section 6.2.3.5, the CEM approach is equivalent to the Boolean algebra performed in PRA software calculations except that Excel spreadsheets are used to perform risk calculations one SU cutset at a time. Then, the Excel spreadsheets are used to sum the contributions for each cutset to obtain the overall MUCDF results.

To produce the cross-unit, dependent failure contributions in the CEM approach, BE coupling factors are assigned, for relevant cutsets, that represent the conditional probability of a failure occurring in Unit 2, given the identical failure occurring in Unit 1. To account for the independent failures between units, the SU CCDP for the MUIE being analyzed is added to the coupling factor. Other cross-unit dependencies (e.g., TDAFW pump failures and dependent HFEs) are handled in the same manner. For those cutsets that do not contain any BEs with the potential for cross-unit dependencies, the MUCDF contribution for that cutset is simply the SUPRA cutset multiplied by the SU CCDP for the MUIE being analyzed, with the SUIEF changed to the MUIEF.

As noted in Section 2.3.4.3 of EPRI's report on MU risk (EPRI, 2021a), "[a] well-known challenge for CCF in [risk-informed decision making] is the potentially scarce actual CCF failures in the operating experience databases" to, for example, support the development of CCF parameters (e.g., alpha factors) for large CCF group sizes. Consequently, there are limitations in the ability of traditional PRA modeling to represent cross-unit or inter-unit CCFs. For these reasons, the L3PRA project team explored multiple options for assigning BE coupling factors for cross-unit CCFs. Appendix H provides more details on how MU CCF coupling factors were developed for the ISR task.

Ultimately, conservative, generic BE coupling factors were assigned to represent MU dependencies (including HFE dependencies). For example, CCF alpha factors used in NRC's SAPHIRE PRA software (INL, 2011) for hypothetical component group sizes of two and four were calculated. MU coupling factors for cross-unit CCFs were selected that were consistent with these calculated alpha factors. Coupling factors for cross-unit human dependencies were developed using the results of the sitewide dependency assessment and analyst judgment. Note, as part of the L3PRA project, options for more refined CCF coupling factors were identified, but not pursued, due to the additional workload given the number of cutsets to be reviewed.

Table 6-1 summarizes the rules for assigning BE coupling factors used for MU CCFs and certain operator actions, including BEs associated with recovery of offsite power.[34] The sitewide dependency assessment performed for the ISR task also informed these assignments. For example, the assessment of potential human and organizational dependencies led to the decision to treat most HFEs in the internal events Level 1 PRA as independent between the two

[34] Although these BEs are quantified using statistical data rather than HRA methods, the L3PRA project's PRA models used HFE labels for these events because the reference plant's PRA used this convention.

units. There were very few cases for which a partial dependency (i.e., use of a 0.1 coupling factor) was used for a cross-unit HFE. Also, the two BEs associated with the recovery of offsite power were assumed to be completely dependent. Appendix H provides more detailed discussions regarding BE coupling factors.

Table 6-1 also shows the cutset types that were identified in the internal events Level 1 PRA for the L3PRA project. The focus of most CEM calculations (except for certain seismic bins) was on cutsets that were assigned a CCF coupling factor. Cutsets that contained only a CCF (beyond the SUIE) were the easiest to address (i.e., MUCDF calculations required the use of only the CCF coupling factor). Two types of CCF coupling factors were considered, CCF1 and CCF2. Cutset type CCF1 was used for most cases involving MU CCFs. Type CCF2 was a special category of CCF that was only used for certain NSCW components that occur in large CCF groups. For example, there are six NSCW pumps and eight NSCW cooling tower fans in each unit at the reference site, leading to a MUCCF group size of 12 and 16, respectively. Given the failure of all six pumps (or eight fans) in one unit from the same common cause, it was judged that the conditional probability of failing all the similar components in the other unit would likely be higher than if the MUCCF group size were smaller. However, as mentioned previously, given that there is limited data of actual CCFs that can support the development of CCF parameters for larger CCF group sizes, it was not practical to calculate an alternate CCF coupling factor. Accordingly, a conservative CCF coupling factor of 1.0 was assigned for type CCF2 (i.e., complete dependence was assumed between units).

Cutsets that contained a CCF in combination with other BEs were given hybrid cutset type labels, such as “CCF1HEP” and “CCF1RANDOM.” Hybrid cutset labels were used in two ways: (1) to identify cutsets that contained both a CCF and a random (or independent) failure, and (2) to identify cutsets that contained two or more dependent BEs (e.g., a CCF and a dependent operator action).

To support MUCDF calculations using the CEM approach, the Excel spreadsheet used to document the review of SU cutsets for potential cross-unit dependencies also documented the assignment of BE coupling factors. For example, for a weather-related LOOP sensitivity case involving additional cutset reviews, the top SU cutset for weather-related LOOPs consists of the following BEs:

- weather-related LOOP IE frequency
- CCF of switchyard AC breakers A and B to open

For this cutset, the SU cutset review identified the CCF as being a potential MU CCF, labeled the cutset as cutset type “CCF1,” and documented a BE coupling factor of 0.2 for the cutset type “CCF1” (per Table 6-1).

Additional iterations of cutset review performed for the weather-related LOOP sensitivity analysis identified and labeled additional BEs for SU cutset types “CCF1” and “CCF2.” Once these BEs were labeled, the Excel spreadsheet was used to document these labels and any assigned coupling factors. Then, the Excel spreadsheet was used to calculate a cutset coupling probability that represents all elements of such hybrid cutsets. Section 6.2.3.4 below discusses these calculations.

Note that treatment of dependencies between reactors that involve HFEs also needs to consider the level of dependence between the units (i.e., full, partial, or none).

Table 6-1 Rules for Assigning BE Coupling Factors for Internal Events Level 1 PRA

Type of Basic Event	Description	Cutset Type	BE Coupling Factor	Notes
CCF	All SU CCFs except those assigned as CCF2 below	CCF1	0.2	Conservative value consistent with hypothetical SAPHIRE-calculated CCF factors
CCF	CCFs associated with certain NSCW components, typically in large CCF groups (e.g., NSCW pumps or cooling tower fans)	CCF2	1.0	Complete dependence assumed due to large CCF group size; judged to be very conservative
CCF	New CCFs associated with TDAFW pump failure to start or run	TDP-CCF1	0.2	Defined as a new, cross-unit CCF for the single TDAFW pump in each unit, forming a new CCF group
HFE	Independent HFEs	HEP	None	HFEs judged to be independent (i.e., failure in Unit 1 does not affect a similar failure in Unit 2)
HFE	Dependent HFEs	HEP	0.1	HFEs judged to have partial dependence between units based on familiarity with reference plant operations and review of cutsets
HFE	Certain HFEs* associated with recovery and restoration of offsite power that are common to both units	N/A	1.0	Complete dependence due to sharing of equipment or resources
Random	All random component failures	RANDOM	None	Independent BEs

* Both BEs defined as HFEs that are related to offsite power recovery are not technically HFEs. Instead, the failure probabilities for these BEs are determined from statistical data. However, the L3PRA project has retained the HFE naming scheme for these events, which originated with the reference plant's PRA.

6.2.3.3.2 Assignment of Cross-Unit Hazard Correlations for External Hazards

One of the challenges for MUCDF calculations for seismic and wind events is that hazard correlations between similar SSCs for Units 1 and 2 are not known. Both IAEA (2021a) and EPRI (2021a) recommend simplified approaches for considering seismic correlations between SSCs in the two units.

In particular, IAEA (2021a) states that "[t]he typical approach is to consider SSCs to be either fully correlated (correlation probability of 1.0) or fully independent (correlation probability 0.0). Full correlation is assumed when a set of SSCs meets all the following conditions:

- they are located in the same building
- they are located on the same level
- they are essentially identical
- they are oriented in the same direction"

EPRI (2021a) describes a similar approach and states that, since "…correlation estimates are based partially on judgment," it is important to have experienced analysts making such judgments. For this reason, the seismic PRA task leader for the L3PRA project selected the seismic hazard correlations used to develop seismic CDF results. For the L3PRA project, the technical lead for the wind PRA judged that no multi-unit hazard correlations should be applied.

For the L3PRA project, MU seismic correlation factors were selected as an extension of those used for the SU seismic PRA (NRC, 2023b). Two generic coupling factors (in addition to those identified previously) were used to develop seismic MUCDF results. The following SU cutset labels, MU hazard correlation factors, and descriptions were defined for these two generic, MU seismic hazard correlations:

- STRUCTURE: Fully correlated (1.0 hazard correlation); used to label two-element SU cutsets that result in direct core damage.

- SEISMIC: Fully correlated (1.0 hazard correlation); used to label SU cutsets that contain only one element related to seismic failures (even if not all IAEA criteria identified above are met).

Assuming full seismic correlation between the two units can be considered a conservative assumption. However, preliminary seismic MUCDF calculations indicated that the extent of conservatism is not very significant and does not justify the level of effort required for a more elaborate analysis.

For SU cutsets in some seismic bins, there were combinations of potential dependencies that were identified. Section 6.2.3.4 discusses the treatment of these combinations in general, while Section 6.2.3.4.3 focuses specifically on seismic events.

6.2.3.4 Calculate Associated Cutset Coupling Probabilities (Step 6)

When traditional PRA software is used, and all cutsets from both units are "ANDed" together, all identified MU dependencies in each cutset need to be addressed or the resulting CDF for the combined MU cutset will be underestimated. Fortunately, PRA software can address these dependencies automatically using post-processing rules and assigned coupling factors. In addition, any independent BEs in an SU cutset are automatically accounted for when generating and quantifying the MU cutsets (which is important to avoid overestimating the CDF of the MU cutsets).

With the CEM approach, identified cross-unit dependencies are addressed on a cutset-by-cutset basis. The CEM approach uses several simplifications, as compared to traditional CDF calculations. Two of these simplifications are (1) often, only one dependency within a cutset is addressed and (2) independent BEs in the cutset are not necessarily addressed for the second unit. These simplifications are implemented based on analyst judgment. Since the CEM approach just applies a coupling factor to the SU cutsets (i.e., it does not carry along the remainder of the cutset from the second unit), both these simplifications can lead to overestimation of MUCDF. If the analyst determines that it is important to address multiple cross-unit dependencies in an SU cutset, then these are treated by calculating a cutset coupling probability (as opposed to just a BE coupling probability). The calculation of a cutset coupling probability reduces the level of overestimation when using the CEM approach, leading to MUCDF results that more closely approximate those that would be obtained using traditional PRA software. However, to manage the level of effort involved, calculation of cutset coupling probabilities in the L3PRA project was limited only to selected cutsets and MUIEs. As discussed in Section 6.2.3.2, additional cutset review can be used to identify SUPRA cutsets that are candidates to have cutset coupling probabilities calculated and applied.

The following sections provide illustrative examples for calculating cutset coupling probabilities for different MUIEs. Section 6.2.3.4.1 provides illustrative examples for a weather-related LOOP sensitivity case. Section 6.2.3.4.2 provides examples for losses of NSCW and Section 6.2.3.4.3 provides examples for seismic events.

6.2.3.4.1 Cutset Coupling Probabilities for Weather-Related LOOPs

Section 12I.1.4 provides additional information regarding how cutset coupling probabilities are calculated and used for generating MUCDF results for weather-related LOOPs. A few examples are provided here (a sampling of the actual cutsets is provided in Table I-8).

First, in the weather-related LOOP sensitivity case cutset number 6, there are three BEs:

- CCF of emergency diesel generators (EDGs) to run
- Two BEs related to offsite power restoration that are common to both units

Per Table 6-1, the following BE coupling factors were assigned to cutset number 6:

- BE coupling factor for CCF of EDGs to run (for cutset type CCF1) = 0.2
- first BE related to offsite power restoration (BE1) = 1.0
- second BE related to offsite power restoration (BE2) = 1.0

In turn, the cutset coupling factor for cutset number 6 (CP_6) was calculated as follows:

$$\begin{aligned} CP_6 &= \text{BE-CCF1} \times \text{BE1} \times \text{BE2} \\ &= 0.2 \times 1.0 \times 1.0 \\ &= 0.2 \end{aligned}$$

Note that, for this particular cutset calculation, the second iteration cutset review, and associated assignment of BE coupling factors, resulted in the same MUCDF contribution that would have been obtained if only the initial cutset review was done (and only the BE-CCF1 coupling factor was used).

For the second example, weather-related LOOP sensitivity case cutset number 27, there are two BEs:

- CCF of EDGs to run
- failure of operators to restore AC power to systems after offsite power is recovered

In this case, the HFE was assessed to be independent (i.e., operator actions associated with this HFE for Unit 1 are independent from those same actions taken for Unit 2). Consequently, instead of a BE coupling factor, the independent human error probability (HEP) assigned to this HFE (i.e., 5.73×0^{-2}) is used to calculate the cutset coupling factor as follows:

$$CP_{27} = \text{BE-CCF1} \times \text{IND-HEP]}$$

$$= 0.2 \times (5.73 \times 10^{-2})$$

$$= 1.15 \times 10^{-2}$$

Consequently, if only the initial cutset review was used for this cutset (i.e., the base case CEM application rather than the weather-related LOOP sensitivity case), the MUCDF contribution for this cutset would be significantly different (i.e., about a factor of 17 higher).

For the third example, weather-related LOOP sensitivity case cutset number 186, there are three BEs:

- CCF of NSCW cooling tower spray valves to open
- failure of operators to recover offsite power
- random BE associated with the fraction of time the NSCW cooling towers are in bypass mode

Per Table 6-1, the following BE coupling factors were assigned to cutset number 186:

- BE coupling factor for CCF of NSCW cooling tower spray valves to open (for cutset type CCF1) = 0.2

- BE related to offsite power restoration (BE1) = 1.0

- random BE independent failure probability (RANDOM) = 9.62×10^{-2}

In turn, the cutset coupling factor for cutset number 186 (CP_{186}) was calculated as follows:

$$CP_{186} = \text{BE-CCF1} \times \text{BE1} \times \text{RANDOM}$$

$$= 0.2 \times 1.0 \times (9.62 \times 10^{-2})$$

$$= 1.92 \times 10^{-2}$$

For the fourth example, weather-related LOOP sensitivity case cutset number 191, there are three BEs:

- CCF of NSCW cooling tower spray valves to open
- failure of operators to restore AC power to systems after offsite power is restored

- random BE associated with the fraction of time the NSCW cooling towers are in spray mode

Per Table 6-1, the following BE coupling factors were assigned to cutset number 191:

- BE coupling factor for CCF of NSCW cooling tower spray valves to open (for cutset type CCF1) = 0.2

- Independent HEP for failure to restore AC power to systems (IND-HEP) = 5.73×10^{-2}

- random BE independent failure probability (RANDOM) = 9.04×10^{-1}

In turn, the cutset coupling factor for cutset number 191 (CP_{191}) was calculated as follows:

$$CP_{191} = \text{BE-CCF1} \times \text{IND-HEP} \times \text{RANDOM}$$

$$= 0.2 \times (5.73 \times 10^{-2}) \times (9.04 \times 10^{-1})$$

$$= 1.04 \times 10^{-2}$$

For the fifth example, weather-related LOOP sensitivity case cutset number 60, there are four BEs:

- CCF of EDGs to run
- TDAFW pump fails to run
- Two BEs related to offsite power restoration that are common to both units

Per Table 6-1, the following BE coupling factors were assigned to cutset number 60:

- BE coupling factor for CCF of EDGs to run (for cutset type CCF1) = 0.2
- BE coupling factor for CCF of TDAFW pumps to run (for cutset type TDP-CCF1) = 0.2
- first BE related to offsite power restoration (BE1) = 1.0
- second BE related to offsite power restoration (BE2) = 1.0

In turn, the cutset coupling factor for cutset number 60 (CP_{60}) was calculated as follows:

$$CP_{60} = \text{BE-CCF1} \times \text{TDP-CCF1} \times \text{BE1} \times \text{BE2}$$

$$= 0.2 \times 0.2 \times 1.0 \times 1.0$$

$$= 0.04$$

For the sixth example, weather-related LOOP sensitivity case cutset number 158, there are two BEs:

- CCF of all six NSCW pumps to start
- random BE associated with RCP seal stage 2 failure

Per Table 6-1, the following BE coupling factors were assigned to cutset number 158:

- BE coupling factor for CCF of all six NSCW pumps to start (for cutset type CCF2) = 1.0

- random BE independent failure probability (RCP) = 0.2

In turn, the cutset coupling factor for cutset number 158 (CP_{158}) was calculated as follows:

$$CP_{158} = \text{BE-CCF2} \times \text{RCP}$$
$$= 1.0 \times 0.2$$
$$= 0.2$$

From the examples provided here, it is seen that the CEM approach could produce different MUCDF estimates for specific SU cutsets that contain both CCFs and other BEs if cutset coupling probabilities are routinely applied. With more complete application of cutset coupling probabilities, the total MUCDF obtained using the CEM approach will more closely match that obtained using traditional PRA software. However, as noted in Section 6.2.3.2, the overall MUCDF estimates for weather-related LOOPs using both a single iteration (i.e., base case CEM application) and multiple iterations (e.g., weather-related LOOP sensitivity case) of cutset review produced almost identical results. As discussed below, there are other cases for which it is more important to address multiple cross-unit dependencies in SU cutsets.

6.2.3.4.2 Cutset Coupling Probabilities for Loss of NSCW

The loss of NSCW event is unique among all MUIEs in four ways: (1) only 50 SU cutsets are needed to represent 95 percent of the SUCDF, (2) all these SU cutsets contain CCFs, (3) all SU cutsets contain at least one other BE (besides the SU CCF), and (4) the SU CCDP is quite high (i.e., 0.25). Furthermore, all the CCFs in these 50 cutsets are for failures of NSCW system equipment in a large CCF group and, therefore, were assigned cutset type “CCF2,” with a coupling factor of 1.0. Consequently, it was necessary to address other BEs in these 50 cutsets to develop MUCDF results that are not overly conservative. The 50 cutsets consist of the following types:

- RCP: The top two SU cutsets for loss of NSCW contain BEs related to RCP seal integrity with a failure probability of 0.2. Since the MU CCF coupling factor for the NSCW CCFs is 1.0, the cutset coupling probability is 0.2 for these two cutsets.

- HEP: The next 16 SU cutsets contain two HFEs that were judged to have cross-unit dependencies: (1) operators fail to establish single pump NSCW operation and (2) operators fail to trip RCPs. For both HFEs, the BE coupling factor was judged to be 0.1. However, only one BE coupling factor was included in the quantification (for simplification). Since the NSCW-related CCF coupling factor is 1.0 (CCF2), the cutset coupling probability is 0.1.

- HEPR: The next 15 SU cutsets contain one of the two HFEs that were judged to have cross-unit dependencies and the BE related to RCP seal integrity failure. With the NSCW-related CCF coupling factor equal to 1.0, the HFE coupling factor equal to 0.1, and the BE failure probability for the RCP seal integrity failure equal to 0.2, the cutset coupling probability is 0.02 (i.e., $1.0 \times 0.1 \times 0.2$).

- OTHER: Cutsets 34, 35, and 48–50 include no additional dependent events and at least one independent (random) BE and cutsets 36-47 contain one of the two HFEs that were judged to have cross-unit dependencies and an independent (random) BE. Due to the low failure probability of the random events, these cutsets would be insignificant contributors to MUCDF even when accounting for the dependencies between units. Therefore, no attempt was made to assign coupling factors to these cutsets.

6.2.3.4.3 Cutset Coupling Probabilities for Seismic Events

There is considerable variation in the types of potential multi-unit dependencies shown in SU cutset results for the different seismic bins. Also, the number of cutsets needed to represent 95 percent of SUCDF varies widely between seismic bins. Similarly, the number and type of hybrid SU cutsets varies for different seismic bins. However, in most cases, cutset coupling probabilities were not calculated. Instead, the seismic PRA team leader reviewed all SU seismic cutsets and made MU seismic correlation factor assignments, based on the generic MU seismic correlation factors discussed in Section 6.2.3.3.2 and the analyst’s experience and understanding of the SU seismic PRA. This involved an iterative process. For this proof-of-concept study, when new hybrid cutset categories were identified or seismic hazard correlation factors assigned, all previously categorized cutsets or assigned factors were not necessarily rigorously reevaluated, particularly where the analyst judged that there would be little impact on the final results.

The SU cutsets for seismic bin 1 have the following characteristics:

- 335 SU cutsets are needed to represent 95 percent of SUCDF.
- Most SU cutsets contain random failures only (except for seismically-induced LOOP).
- 42 SU cutsets contain a single CCF of either type “CCF1” or “CCF2.”
- 11 SU cutsets are labelled as hybrid cutsets but no cutset coupling probabilities were calculated for any of these cutsets (i.e., a single BE coupling factor was used to estimate MUCDF):
    - 2 SU cutsets are labelled “CCF1HEP” with the operator action judged to be independent.
    - 1 SU cutset is labelled "HEPS” with two HFEs in the cutset.
    - 8 SU cutsets are labelled "HEPSR” with two HFEs and a random failure in each cutset.

In sharp contrast, only one cutset was needed to represent SUCDF for seismic bin 8, and this cutset contained only two elements: the IE frequency and a seismic hazard correlation of 1.0. Although more SU cutsets were needed to estimate MUCDF for seismic bin 7 (i.e., 16 SU cutsets), the SU cutsets for bins 7 and 8 were similar in that none of these cutsets were given hybrid labels (and the MU CCDP was assumed to be 1.0). SU cutsets for the other seismic bins are more complicated, including more hybrid SU cutsets.

Table 6-2 shows the different hybrid SU cutset labels that were used to assign MU seismic hazard correlation factors. This table also provides a description of the SU cutset elements associated with the labels, the recommended hazard correlation factors, and the actual occurrences and assigned factors by seismic bin. In summary, Table 6-2 also shows:

- By the lack of entries, there were no hybrid SU cutsets for seismic bins 1, 7 and 8.
- As shown by the gray shading, there are only two types of hybrid SU cutsets for which a cutset coupling probability was calculated (although the calculation was trivial since the MU seismic hazard correlation was equal to 1.0):
    - CCF1-SEISL
    - HEP2L[35]
- As shown by the blue shading, there are four types of hybrid SU cutsets which always have an MU seismic correlation factor of 1.0:
    - SEISMIC2
    - SEISMIC2L
    - SEISMIC3
    - SEISMIC3L
- All other hybrid SU cutsets were recommended to be assigned a MU seismic correlation factor between 0.01 and 0.1. (Note, a default seismic correlation factor of 0.1 was used initially. Due to the greater potential impact on MUCDF, the analyst judged it to be more appropriate to assign a less conservative factor of 0.01 for the higher seismic bins [i.e., bins 5 and 6].)
- OTHER: Various combinations of seismic, common-cause, and random failures (either equipment failures or HFEs).

[35] Based on analyst judgment a single factor of 0.1 was used to collectively account for both HFEs in the cutset.

Table 6-2 Hybrid, SU Cutsets Labels, Description, Hazard Correlations, and Occurrences

Hybrid Cutset Label	Description	Recommended Seismic Hazard Correlation	Number of Occurrences (Assigned Seismic Hazard Correlation)
CCF1-SEISL	Seismically induced LOOP and a CCF1	0.2 (cutset coupling probability calculated as 0.2 x 1.0)	Bin 3: 5 Bin 4: 3
HEP2L	Seismically induced LOOP and two partially dependent HFEs	0.1 (cutset coupling probability calculated as 0.1 x 1.0)	Bin 3: 1
SEISMIC2	2 seismic BEs in the same cutset	1.00	Bin 5: 12 Bin 6: 10
SEISMIC2L	2 seismic BEs (1 is a seismically induced LOOP)	1.00	Bin 2: 15 Bin 3: 14 Bin 4: 16
SEISMIC2LHEP	2 seismic BEs (1 is a seismically induced LOOP) and an HFE	0.01–0.10	Bin 3: 1 (0.10) Bin 4: 1 (0.10)
SEISMIC2LR	2 seismic BEs (1 is a seismically induced LOOP) and an RCP seal failure BE	0.01–0.10	Bin 2: 2 (0.10) Bin 3: 3 (0.10) Bin 4: 2 (0.10)
SEISMIC2R	2 seismic BEs and an RCP seal failure BE	0.01–0.10	Bin 5: 1 (0.01) Bin 6: 1 (0.01)
SEISMIC2U1	2 seismic BEs (1 is seismically uncorrelated)	0.01–0.10	Bin 5: 15 (0.01) Bin 6: 13 (0.01)
SEISMIC2U1L	2 seismic BEs (1 is seismically uncorrelated and 1 is a seismically induced LOOP)	0.01–0.10	Bin 3: 1 (0.10)
SEISMIC2U2	2 seismic BEs (2 are seismically uncorrelated)	0.01–0.10	Bin 5: 5 (0.01) Bin 6: 6 (0.01)
SEISMIC3	3 seismic BEs	1.00	Bin 5: 1 Bin 6: 4
SEISMIC3L	3 seismic BEs (1 is a seismically induced LOOP)	1.00	Bin 3: 4 Bin 4: 8

Table 6-2 Hybrid, SU Cutsets Labels, Description, Hazard Correlations, and Occurrences (cont.)

Hybrid Cutset Label	Description	Recommended Seismic Hazard Correlation	Number of Occurrences (Assigned Seismic Hazard Correlation)
SEISMIC3R	3 seismic BEs and an RCP seal failure BE	0.01–0.10	Bin 4: 1 (0.10) Bin 5: 3 (0.01) Bin 6: 5 (0.01)
SEISMIC3RL	3 seismic BEs (1 is a seismically induced LOOP) and an RCP seal failure BE	0.01–0.10	Bin 3: 1 (0.10) Bin 4: 1 (0.10)
SEISMIC3U1	3 seismic BEs (1 is uncorrelated)	0.01–0.10	Bin 5: 2 (0.01) Bin 6: 3 (0.01)
SEISMIC3U1L	3 seismic BEs (1 is uncorrelated, and 1 is a seismically-induced LOOP)	0.01–0.10	Bin 3: 8 (0.10) Bin 4: 10 (0.10)
SEISMIC3U2L	3 seismic BEs (2 are uncorrelated and 1 is a seismically-induced LOOP)	0.01–0.10	Bin 3: 3 (0.10) Bin 4: 3 (0.10)
SEISMIC4U1R	4 seismic BEs (1 is uncorrelated) and an RCP seal failure BE	0.01–0.10	Bin 6: 2 (0.01)
SEISMICR	1 seismic BE and an RCP seal failure BE	0.01–0.10	Bin 2: 2 (0.10) Bin 3: 1 (0.10) Bin 4: 2 (0.10) Bin 5: 1 (0.01) Bin 6: 1 (0.01)
OTHER	Various combinations of seismic, common-cause, and random failures		Bin 2: 4 (0.10) Bin 3: 5 (0.10) Bin 4: 6 (0.10) Bin 5: 4 (0.01) Bin 6: 1 (0.01)

6.2.3.5 *Calculate MUCDFs (Step 7)*

As mentioned previously, to support MUCDF calculations using the CEM approach, the Excel spreadsheet used to document the review of SU cutsets for potential cross-unit dependencies also documented the assignment of BE and cutset coupling factors. A separate, but linked, spreadsheet within the Excel notebook was then used to perform the MUCDF calculations that are described below. The calculation of MUCDF using the CEM involves only a few simple steps:

1. For each SU cutset selected to represent 95 percent of the SUCDF, represent two contributions to MUCDF via Boolean algebra:

    a. the cross-unit, dependent cutset (using the BE or cutset coupling probability) **IF** a potential cross-unit CCF (or seismic failure) was identified for the specific cutset

    b. “random” or independent MU cutsets (using the SU CCDP)

2. Sum results for each SU cutset.

3. Apply a scale-up factor to estimate total MUCDF.

The following terms from the SUPRA are used in the CEM approach to estimate MUCDF:

U1IEF:	Unit 1 initiating event frequency
U1CDF (total):	Total single-unit CDF (SUCDF) calculated in L3PRA project (for a specific IE)
U1-CCDP:	Unit 1 conditional core damage probability (for a specific IE)
U1CDF (95%):	Unit 1 SUCDF for the N cutsets selected for CEM (where N is the number of cutsets that represent 95 percent of total CDF for a specific IE)
$U1\text{-}CDF_i$:	Unit 1 SUCDF for cutset i out of N cutsets

For the reference site, Units 1 and 2 are essentially identical for the MUIEs of interest. Consequently, “U2” can be substituted for “U1” for all the terms above. (Also, for this discussion, U1CDF and SUCDF have the same meaning.)

Other terms needed for the CEM to calculate MUCDF include:

$U1\text{-}CCDP_i$	Single unit CCDP for cutset i only (i.e., failure probability for all BEs in cutset i)
MUIEF:	Multi-unit initiating event frequency (also sitewide IE frequency)
MUCDF:	Multi-unit CDF (calculated with CEM)
$MUCDF_i$:	MU core damage frequency for cutset i of N cutsets

Scale-up factor: Adjustment of CEM MUCDF to account for use of only 95 percent of SUCDF

$BE\text{-}CF_i$: BE coupling factor for cutset i (assigned)

$CS\text{-}CP_i$: Cutset coupling probability for cutset i (calculated), if needed

Using these terms, the MUCDF for cutset i was calculated as the following (for all cases that did not require calculation of a cutset coupling probability):

$$MUCDF_i = U1CDF_i \times (MUIEF/U1IEF) \times [BE\text{-}CF_i + U1\text{-}CCDP - (BE\text{-}CF_i \times U1\text{-}CCDP)]$$

Other terms or contributions in the above equation that also need explanation include:

$U1CDF_i =$ $U1IEF \times U1\text{-}CCDP_i$

$U1CDF_i \times MUIEF/U1IEF =$ $MUIEF \times U1\text{-}CCDP_i$

$U1CDF_i \times (MUIEF/U1IEF) \times BE\text{-}CF_i$: Dependent contribution to $MUCDF_i$

$U1CDF_i \times (MUIEF/U1IEF) \times [U1\text{-}CCDP]$: Independent contribution to $MUCDF_i$

$BE\text{-}CF_i \times U1\text{-}CCDP$: Term needed for rare events approximation in Boolean algebra

After the CEM approach is applied to each of the selected Unit 1 minimal cutsets, the MUCDF contributions from each cutset are summed to obtain the overall MUCDF. This MUCDF result is associated with the SU cutsets identified as contributing to 95 percent of the SUCDF for the specific IE. To obtain an estimate of the “final MUCDF,” a scale-up factor is applied. Application of a scale-up factor is especially important if the CDF for selected SU cutsets is significantly different than the total SUCDF. Use of the scale-up factor assumes that the remaining cutsets that are not examined (the residue) have the same characteristics as the top 95 percent. The scale-up factor is calculated as:

$$\text{Scale-up} = U1CDF\ [total] / U1CDF\ [95\%]$$

Two additional terms related to the two extreme MUCDF cases for a two-unit site also were used in the ISR task:

MAX-MUCDF: Maximum possible MUCDF, assuming complete dependence between the two units

MIN-MUCDF: Minimum possible MUCDF, assuming complete independence between the two units

Note that if MUIEF is equal to the SUIEF (e.g., for seismic events), then the above equations are further simplified as:

$$MUCDF_i = U1CDF_i \times [BE\text{-}CF_i + U1\text{-}CCDP - (BE\text{-}CF_i \times U1\text{-}CCDP)]$$

Section 12I.1 provides illustrative examples of cutset reviews, assignment of cutset types and BE coupling factors, and MUCDF calculations.

6.3 <u>Summary of Results</u>

This section summarizes the MUCDF results produced by the ISR task. Section 12I.3 provides more detailed results.

6.3.1 Summary of MUCDF Results for the Base Case

Table 6-3 shows all the MUCDF results that are based on the Unit 1 “base case” (i.e., Circa 2012) PRAs. Shading is used in this table to distinguish between the different types of MUIEs (e.g., LOOPs are shaded with blue, loss of NSCW is shaded with yellow, and fires are shaded with orange). In addition, the four seismic bins that are the largest contributors to MUCDF (i.e., seismic bins 3, 4, 5, and 6) are shaded with purple. The other four seismic bins are shaded with gray.

Several insights can be obtained from Table 6-3, such as:

- The contribution to total MUCDF from all LOOP events combined is about 14 percent, with grid-related LOOPs contributing the most (about 8 percent of total MUCDF).
- Seismic events contribute over half of the total MUCDF (about 53 percent) with:
    - Bins 3 through 6 contributing nearly 94 percent of the total seismic MUCDF.
    - Bins 4 through 6 contributing over 80 percent of the total seismic MUCDF, in nearly equal shares.
- The contribution to total MUCDF from the loss of NCSW initiating event is about 25 percent.
- The contribution to total MUCDF from wind-related events is 6 percent.
- The contribution to total MUCDF from all four fire scenarios is about 2 percent.

Since the estimated contribution to MUCDF from the loss of NSCW initiating event is significant (i.e., 25 percent), and perhaps unexpected, some additional information on this contribution is provided. In particular, two modeling assumptions used to estimate MUCDF for LONSCW contribute to this result:

- The LONSCW initiating event frequency was modeled in the L3PRA by a fault tree. This FT is dominated by CCFs of NSCW pumps, which are assumed to affect both units (i.e., these BEs were assigned a conservative coupling factor of 1.0). This assumption is consistent with the treatment of these types of failures when modeled as subsequent BEs that follow other MUIEs (i.e., assigned as cutset type CCF2). If a less conservative coupling factor is assumed for these events, then the estimated MUCDF would be correspondingly reduced.
- As noted in Section 6.2.3.4.2, there are a considerable number of significant Unit 1 loss of NSCW cutsets that have failures of operator actions that were judged to have cross-unit dependencies. These actions are associated with restoration of NSCW pumps and tripping RCPs. The cutsets that contained these operator actions were assigned as

cutset type HEP, with a BE coupling factor of 0.1, as shown in Table 6-1. This BE coupling factor assignment implies a high correlation between the failure of these actions in both units. Assuming a lower cross-unit correlation for these actions could reduce the estimated MUCDF up to 20 percent.

In addition, the following should be noted:

- The MUCDF results for all LOOP categories shown in Table 6-3 were calculated in the same way (i.e., considering MU CCFs as the only source of MU dependencies, besides the MUIE). As noted previously, additional iterations of review for cutsets with MU CCFs were performed for weather-related LOOPs. These additional cutset reviews also led to calculations to address MU dependencies for restoration of offsite power and independent random events (both operator actions and equipment failures) for the applicable cutsets. However, the calculated MUCDF for this demonstration of the CEM approach (i.e., 4.35×10^{-7} per reactor-critical-year) was not much different than the result produced without these refinements (i.e., 4.47×10^{-7}per reactor-critical-year). Consequently, the weather-related LOOP MUCDF result reported here is for the simpler case that addressed MU CCFs only (i.e., after only a single iteration of cutset review). This choice also simplifies the comparison of results because the basis of the calculations is then the same for all categories of LOOP.

- All four fire scenarios that contribute to MUCDF were already modeled in the SU fire PRA. Consequently, no new modeling was needed.

- The MUCDF results for seismic events were developed through treatment of MU CCFs and seismic hazard correlations in application of the CEM approach. More insights on these results are given below in Section 6.3.2.

- The overall MUCDF results for winds events were developed by considering 12 different wind scenarios. MU CCFs were the only MU dependencies addressed; no additional hazard correlations were applied. Section 12I.3.1.4 provides more details on these calculations.

Table 6-3 MUCDF Estimates – Base Case

	Scenario Name	Scenario Description	MU Scenario Characteristics	MUIEF (/rcy)	MUCDF (/rcy)	% MUCDF (/rcy)
1	MU-IE-LOOPGR	Grid-related LOOP	SBO and AC power recovery failure	6.15E-03	1.00E-06	7.7%
2	MU-IE-LOOPPC	Plant-centered LOOP	SBO and AC power recovery failure	1.07E-04	1.43E-08	0.1%
3	MU-IE-LOOPSC	Switchyard-centered LOOP	SBO and AC power recovery failure	2.80E-03	3.56E-07	2.7%
4	MU-IE-LOOPWR	Weather-related LOOP	SBO and AC power recovery failure	2.44E-03	4.47E-07	3.4%
5	MU-LONSCW	Loss of NSCW	Loss of NSCW in both units	3.47E-05	3.23E-06	24.8%
6	MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned with CCDP =1	1.47E-07	1.47E-07	1.1%
7	MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2	at least MU LOOP (assumed)	3.42E-02	2.28E-08	0.2%
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.5%
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.5%
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic BIN-1	1.64E-03	8.03E-08	0.6%
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic BIN-2	2.19E-04	1.24E-07	1.0%
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic BIN-3	4.79E-05	8.24E-07	6.3%
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic BIN-4	1.34E-05	1.85E-06	14.2%
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic BIN-5	4.26E-06	2.02E-06	15.6%
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic BIN-6	1.92E-06	1.72E-06	13.3%
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2-unit SBO and Major structural damage (EQK-BIN7) with CCDP =1	2.48E-07	2.34E-07	1.8%
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and Major structural damage (EQK-BIN8) with CCDP = 1	2.32E-09	2.32E-09	<0.1%
18	MU-IE-WIND-1	SBO and SSC wind damage	SBO and WIND damage to SSCs	8.89E-03	7.93E-07	6.0%
			Total =	7.47E-02	1.30E-05	100.0%

rcy – reactor-critical-year

6.3.2 Comparison of MUCDF Results to SUCDF Results for the Base Case

Table 6-4 provides further insights through comparison of the MUCDF results to the SUCDF results for the base (Circa 2012) case. Table 6-4 shows that, overall, total MUCDF is about 10 percent of SUCDF. Table 6-4 also shows that:

- For seismic events, the ratio of MUCDF to SUCDF increases with increasing seismic bin number, exceeding 50 percent for seismic bins 3–8.
- The ratio of MUCDF to SUCDF is also significant for loss of NSCW (i.e., about 37 percent).
- The ratio for all other MUIEs is significantly less than 10 percent (except for MCR abandonment due to fire, which is assumed to be completely dependent between units, i.e., ratio of 1.0).

Additional insights from Table 6-4, pertaining to individual MUIEs, are as follows:

- Seismic events:
    - Seismic bins 5–8: The MUCDF is identical, or nearly identical, to the U1CDF (i.e., between 90 and 100 percent of U1CDF) due to the complete, or nearly complete, dependence between the units as a result of the MU seismic hazard correlations used in the MUCDF calculations. For seismic bin 6 (which is used later for the illustrative multi-source scenario), cutsets containing MU seismic hazard correlations (e.g., MU coupling factors) account for 97 percent of MUCDF.
    - Seismic bins 3 and 4: The MUCDF is about 50 percent and 75 percent of the U1CDF for bins 3 and 4, respectively. For both bins, MU seismic hazard correlations dominate the MUCDF results, but bin 4 has more cutsets with MU dependencies than bin 3 does (i.e., 68 versus 59 SU cutsets with MUCDF contributions).
    - Seismic bin 2: The MUCDF is about 10 percent of the U1CDF, and the largest contributing dependencies arise from cross-unit CCFs (though there are also some significant MUCDF contributions from cutsets that have MU seismic hazard correlations applied).
    - Seismic bin 1: The MUCDF is 6 percent of the U1CDF, and the principal dependencies that drive these results are cross-unit CCFs (similar to LOOPs). These results imply that the effect of MU CCFs on MUCDF is smaller than the effect from MU seismic hazard correlations.

- Wind:
    - The MUCDF for the all-encompassing wind IE is about 0.6 percent of the U1CDF. While the bulk of the total wind MUCDF comes from cross-unit CCFs,[36] these contributions are small compared to the U1CDF.
- Fires:
    - As mentioned previously, main control room abandonment scenarios have been assumed to affect both units identically so, as expected, the MUCDF and U1CDF are identical. This scenario was modeled in the SUPRA and no further modeling or calculations were needed for the ISR task.
    - MUCDF contributions from fires in shared areas are only about 2 percent of the U1CDF. These scenarios result in MU-LOOP events due to the fire location. There are several physical analysis units (PAUs) addressed by the SU fire PRA that are shared between units, including the switchyards, the general outside areas, and several other general areas in the auxiliary building, control building, and fuel handling building. A review of the results for these shared locations indicates only the high and low switchyard and general yard areas have a noticeable contribution to MU risk. Switchyard fires may damage equipment or cables for offsite power for one or both units.
    - MUCDF contributions from fires that cascade from one unit to the other (i.e., Unit 2 to Unit 1 or Unit 1 to Unit 2) have the smallest ratio of MUCDF to U1CDF (i.e., 0.2 percent of U1CDF). Because Units 1 and 2 are considered identical, the MUCDF contributions are the same for both scenarios, but only fires cascading from Unit 2 to Unit 1 were modeled in the SU (Unit 1) fire PRA.
- Loss of NSCW:
    - The MUCDF for this MUIE is calculated to be 37 percent of the U1CDF. As explained in the previous section, potentially conservative cross-unit CCF factors and partially dependent operator actions associated with starting NSCW pumps and tripping RCPs are the dominant drivers for the MUCDF results for this MUIE.
- LOOPs:
    - Plant-centered LOOPs: The MUCDF for plant-centered LOOPs is less than 1 percent of the U1CDF, making it the smallest contributor to MUCDF after fires that cascade from one unit to the other. The principal contributors to MUCDF for plant-centered LOOPs are MU CCFs.
    - All other LOOPs: MUCDFs of all other LOOPs are generally around 5 percent of the respective U1CDF. The principal contributors to MUCDF for these LOOPs are also MU CCFs.

[36] Recall from Section 6.2.3.3.2 that for the L3PRA project, the technical lead for the wind PRA judged that no multi-unit hazard correlations should be applied.

Table 6-4 Comparison of Single Unit IEFs and CDF with MUIEFs and MUCDF

	Scenario Name	Scenario Description	U1IEF (/rcy)	MUIEF (/rcy)	MUIEF/ U1IEF	U1CDF (/rcy)	MUCDF (/rcy)	MUCDF/ U1CDF
1	MU-IE-LOOPGR	Grid-Related LOOP	1.23E-02	6.15E-03	0.500	1.83E-05	1.00E-06	0.055
2	MU-IE-LOOPPC	Plant-Centered LOOP	1.93E-03	1.07E-04	0.056	1.91E-06	1.43E-08	0.007
3	MU-IE-LOOPSC	Switchyard-Centered LOOP	1.04E-02	2.80E-03	0.269	1.04E-05	3.56E-07	0.036
4	MU-IE-LOOPWR	Weather-Related LOOP	3.91E-03	2.44E-03	0.625	9.02E-06	4.47E-07	0.049
5	MU-LONSCW	Loss of NSCW	3.47E-05	3.47E-05		8.76E-06	3.23E-06	0.369
6	MU-IE-FRI-1	MCR abandonment due to fire	1.50E-07	1.47E-07	1.0	1.47E-07	1.47E-07	1.0
7	MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2	3.40E-02	3.42E-02	1.0	1.00E-05	2.28E-08	0.023
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)		9.08E-03	1.0		6.59E-08	-
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	9.08E-03	9.08E-03	1.0	4.23E-05	6.59E-08	0.002
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	1.64E-03	1.64E-03	1.0	1.30E-06	8.03E-08	0.062
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2.19E-04	2.19E-04	1.0	1.22E-06	1.24E-07	0.102
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	4.79E-05	4.79E-05	1.0	1.62E-06	8.24E-07	0.509
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	1.34E-05	1.34E-05	1.0	2.43E-06	1.85E-06	0.761
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	4.26E-06	4.26E-06	1.0	2.24E-06	2.02E-06	0.902
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	1.92E-06	1.92E-06	1.0	1.75E-06	1.72E-06	0.983
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2.48E-07	2.48E-07	1.0	2.34E-07	2.34E-07	1.000
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2.32E-09	2.32E-09	1.0	2.32E-09	2.32E-09	1.000
18	MU-IE-WIND-1	SBO and SSC wind damage	8.89E-03	8.89E-03	1.0	1.38E-05	7.93E-07	0.006
	Total		8.24E-2	7.47E-02		**1.25E-4**	**1.30E-05**	0.104

6.3.3 Illustrative Results for Maximum and Minimum MUCDFs

Section 6.2.3.5 introduced the terms for the maximum and minimum possible MUCDF. The conservatism (or lack thereof) in MUCDF estimations can be assessed through calculation of the maximum and minimum possible MUCDF. Calculation of the maximum and minimum possible MUCDFs also provides a self-check on the appropriateness of the MUCDF calculations (i.e., the calculated MUCDF should be no smaller than the minimum MUCDF and no greater than the maximum MUCDF).

Using the terms defined in Section 6.2.3.5, these two calculations can be performed with the following equations, where Unit 1 and Unit 2 have identical cutsets and CDF results:

$$\text{MAX-MUCDF} = \text{MUIEF} \times \text{U1-CCDP} \times 1.0$$

$$\begin{aligned}\text{MIN-MUCDF} &= \text{MUIEF} \times \text{U1-CCDP} \times \text{U2-CCDP} \\ &= \text{MUIEF} \times (\text{U1-CCDP})^2\end{aligned}$$

In the previous section, seismic bins 6, 7, and 8 were identified as having an MUCDF equal (or almost completely equal) to the Unit 1 (or SU) CDF. These results are a special case in which the MUIE and SUIE frequencies are identical and there is complete dependence between the units (resulting from the MU seismic hazard correlations). In other words, the calculated MUCDF for seismic bins 6, 7, and 8 is equivalent to the MAX-MUCDF for these seismic bins. The same is true for the dual-unit, main control room abandonment scenario for fire events (i.e., the calculated MUCDF is equal to the MAX-MUCDF).

Table 6-5 shows MUCDF results calculated with the CEM approach, MIN-MUCDF, and MAX-MUCDF for six scenarios—all four LOOPs, loss of NSCW, and wind events. Note that the MUIEFs for the LOOPs are all smaller than the corresponding U1IEFs, while the MUIEFs for the loss of NSCW and wind events are equal to the U1IEFs.

Based on the information reported in Table 6-5, Figure 6-1 compares the MUCDF results calculated with the CEM approach with the maximum and minimum MUCDF results for the base (Circa 2012) case. Results for the four LOOPs (for which the MUIEF differs from the U1IEF) are shown, as well as those results for the loss of NSCW. The following conventions are used in Figure 6-1:

- The “X” shown at the bottom of the vertical line for each MUIE represents the MIN-MUCDF.
- The orange circles represent the MUCDF estimated with the CEM approach.
- The “-“ shown at the top of the vertical line for each MUIE represents the MAX-MUCDF.

The simplest insights that can be gained from Figure 6-1 arise from comparisons of estimated MUCDF with MIN-MUCDF and MAX-MUCDF. For example, a CEM result that is “too close to” the minimum or maximum MUCDF might indicate the need for a re-check of how MU dependencies were treated. Table 6-5 and Figure 6-1 show, for all four LOOPs:

- a factor of approximately 35–140 between estimated MUCDF and MIN-MUCDF
- a factor of approximately 7–13 between estimated MUCDF and MAX-MUCDF

On the other hand, Table 6-5 and Figure 6-1 show that the values for estimated MUCDF, MIN-MUCDF and MAX-MUCDF for loss of NSCW are very close together (i.e., a factor of approximately 4 between MIN-MUCDF and MAX-MUCDF). For this case, the project team is well aware of the conservative assumptions made regarding MU dependencies for the NSCW system. However, the project team thinks that these conservatisms are justified for the purposes of the L3PRA project and the associated ISR task, especially given the lack of applicable data for calculating CCF parameters for large CCF component groups.

Table 6-5 Comparison of MUCDF, MIN-MUCDF, and MAX-MUCDF for LOOPs, LONSCW, and Wind Events for Base Case (Circa 2012)

Scenario Name	Scenario Description	U1IEF (/rcy)	MUIEF (/rcy)	MU Multiplier	U1CDF (/rcy)	MUCDF (/rcy)	MIN-MUCDF (/rcy)	MAX-MUCDF (/rcy)
MU-IE-LOOPGR	Grid-Related LOOP	1.23E-02	6.15E-03	0.500	1.83E-05	1.00E-06	1.37E-8	9.17E-6
MU-IE-LOOPPC	Plant-Centered LOOP	1.93E-03	1.07E-04	0.056	1.91E-06	1.43E-08	1.05E-10	1.06E-7
MU-IE-LOOPSC	Switchyard-Centered LOOP	1.04E-02	2.80E-03	0.269	1.04E-05	3.56E-07	2.77E-9	2.78E-6
MU-IE-LOOPWR	Weather-Related LOOP	3.91E-03	2.44E-03	0.625	9.02E-06	4.47E-07	1.30E-8	5.63E-6
MU-LONSCW	Loss of NSCW	3.47E-05	3.47E-05	1.0	8.76E-06	3.23E-06	2.17E-6	8.74E-6
MU-IE-WIND-1	SBO and SSC wind damage	8.89E-03	8.89E-03	1.0	1.38E-05	7.93E-07	8.99E-08	1.38E-05

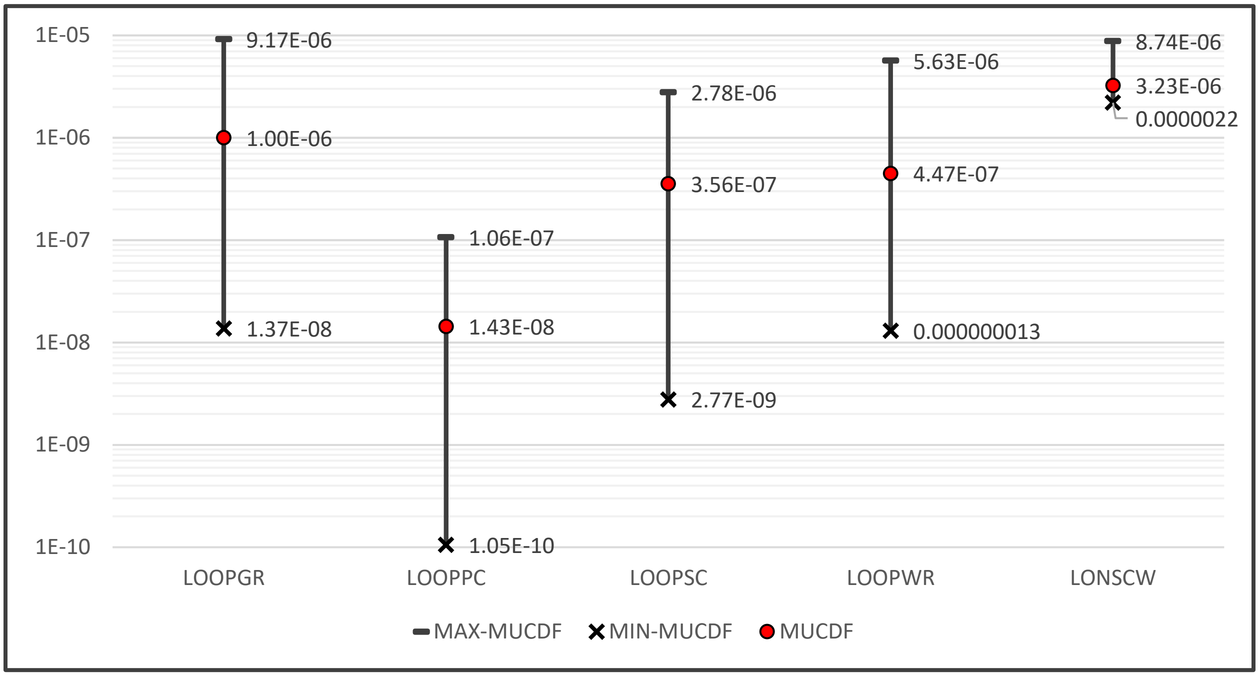

Figure 6-1 Comparison of MUCDF Estimates to Maximum and Minimum MUCDF for LOOPs, and Loss of NSCW (Circa 2012 – Base Case)

6.3.4 Illustrative Examples of MUCDF Contributions from Different Types of MU Dependencies

The MUCDF results developed for the base case were examined further to identify which types of MU dependencies were important contributors to MUCDF.[37] For five illustrative examples, Table 6-6 shows MUIEF, U1CDF, MUCDF and percent contribution to MUCDF from MU dependencies.

The five MUIEs shown in Table 6-6 are:

- grid-related LOOPs
- weather-related LOOPs
- seismic bin 2
- seismic bin 6
- wind events

For illustrative purposes, the data in Table 6-6 are shown graphically in Figure 6-2, Figure 6-3, and Figure 6-4 for grid-related LOOPs (although the results for weather-related LOOPs are essentially identical), seismic bin 2, and seismic bin 6, respectively.

Both Table 6-6 and Figure 6-2 show that MU CCFs contribute essentially the entire MUCDF for both LOOPs. For both types of LOOP, MU CCFs of cutset type “CCF1” are the predominant contributors (i.e., about 86 percent), while MU CCFs of type “CCF2” contribute about 14 percent. These results are expected since MU CCFs were the MU dependencies that the lead

[37] Because MUCDF cutsets are not developed when the CEM approach is applied, it is not possible to use traditional importance measures for the ISR task.

analyst primarily focused on, based on their familiarity with the SU model cutsets and the results of the sitewide dependency assessment.

The results for wind events shown in Table 6-6 are fairly similar to those for the two LOOPs, although random (or independent) failures make a significant contribution to MUCDF for wind events (approximately 11 percent).

Both grid-related and weather-related LOOPs have SU cutsets with BEs related to recovery of offsite power that were identified as MU dependencies. However, the dominant contributors to MUCDF that contain these BEs also contain MU CCFs, so the impact of the MU dependencies for offsite power recovery cannot be separated out. As shown in Section 12I.1.4 , contributions from these BEs in weather-related LOOP cutsets that contain random (or independent) failures, instead of CCFs, contribute very little to MUCDF. Since grid-related and weather-related LOOP SU cutsets are very similar, a similar insight would be expected for grid-related LOOPs.

Figure 6-3 and Figure 6-4 (and the underlying results shown in Table 6-6) show very different contributions from MU dependencies for seismic bins 2 and 6. For example, while the dominant contributor to MUCDF for both seismic bins is from seismic correlations, the seismic correlation contribution for seismic bin 6 (e.g., 97 percent) is significantly larger than for seismic bin 2 (e.g., about 64 percent), especially when considering the seismic correlations for structural failures. Also, MUCDF for seismic bin 2 has a significant contribution from MU CCFs, while there are no such contributions for seismic bin 6.

Table 6-6 Illustrative MUCDF Results and Contributions from Different Types of MU Dependencies

Scenario Name	Scenario Description	MUIEF (/rcy)	U1CDF (/rcy)	MUCDF (/rcy)	% CCF1	% CCF2	% Random	% Seismic	% Structure	% Other
MU-IE-LOOPGR	Grid-related LOOP	6.15E-03	1.83E-05	1.00E-06	85.5	14.3	0.1	N/A	N/A	N/A
MU-IE-LOOPWR	Weather-related LOOP	2.44E-03	9.02E-06	4.47E-07	85.6	14.1	0.2	N/A	N/A	N/A
MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2.19E-04	1.22E-06	1.24E-07	27.1	4.3	4.4	59.9	3.6	0.7
MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	1.92E-06	1.75E-06	1.72E-06	0	0	2.3	73.7	23.5	0.5
MU-IE-WIND-1	SBO and SSC wind damage	8.89E-03	1.38E-05	7.93E-07	76.4	12.2	11.4	N/A	N/A	N/A

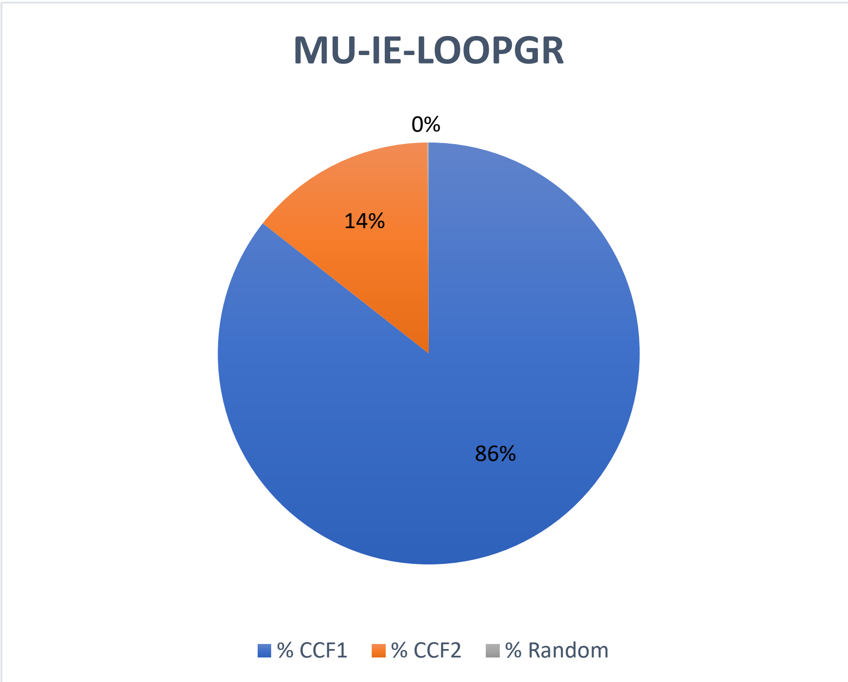

Figure 6-2 Sitewide Dependency Contributions to Grid-Related LOOPs

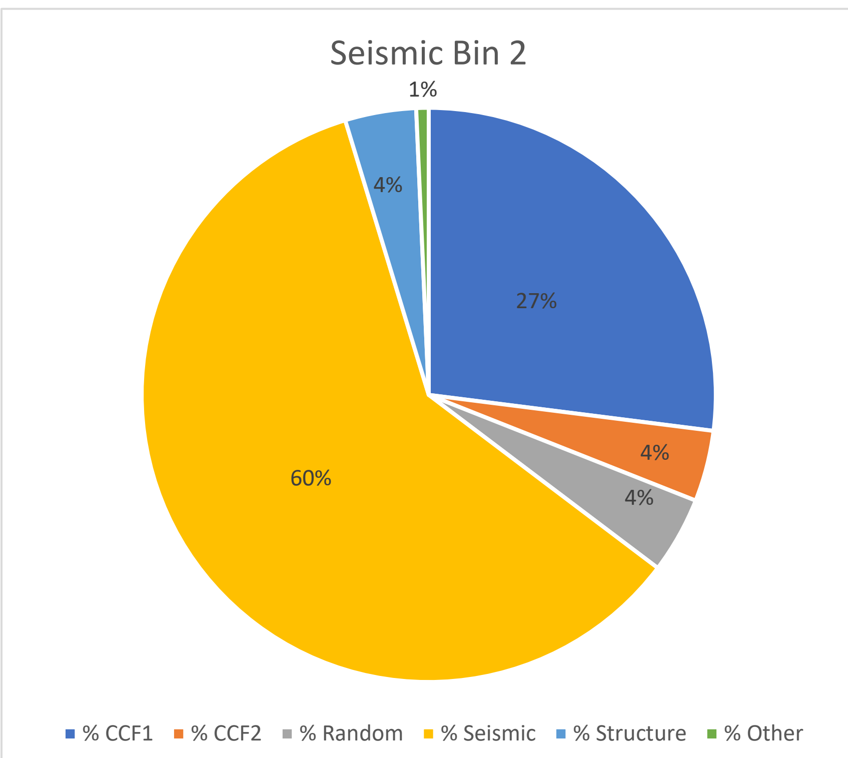

Figure 6-3 Sitewide Dependency Contributions to Seismic Bin 2

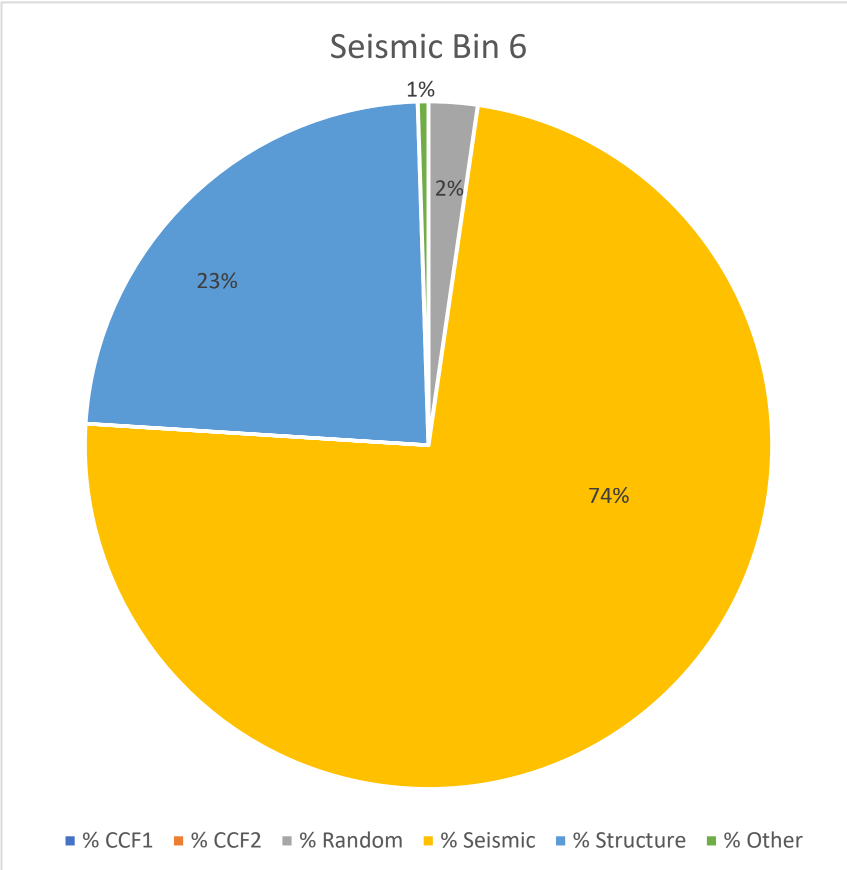

Figure 6-4 Sitewide Dependency Contributions to Seismic Bin 6

6.3.5 Summary of MUCDF Results for the FLEX Sensitivity Case

As mentioned previously, the base case for the L3PRA project is based on 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. For the ISR task, FLEX sensitivity case MUCDF calculations were performed for LOOPs, seismic events, and losses of NSCW. The FLEX sensitivity case included credit for declaration of extended loss of AC power (ELAP) and implementation of FLEX strategies and extended TDAFW operation in the absence of all installed AC and DC power, as well as for the new RCP shutdown seals. Otherwise, the same assumptions were used as for the base case.

Table 6-7 shows MUCDF results for the base and FLEX cases (although FLEX cases have not been performed for fire scenarios). As can be seen in Table 6-7, the FLEX sensitivity case results in approximately a 50 percent reduction in total MUCDF.[38] Also, like the SUCDF results,

[38] Some reclassification of coupling factors was performed on the seismic cutsets for the base case, and this is reflected in the results provided in this report. The modified coupling factors had very minimal impact on the reported results. A similar reclassification was not performed for the coupling factors for the FLEX sensitivity case, though this is not expected to have any meaningful impact on the results and insights for the FLEX sensitivity case.

the MUCDF results show that modifications associated with the FLEX sensitivity case are most effective for LOOPs, losses of NSCW, and wind events. MUCDF for LOOPs and wind events (which are dominated by wind-related LOOPs) is reduced substantially because the FLEX strategies are targeted at ELAPs. Most loss of NSCW sequences involve an RCP seal loss-of-coolant accident (LOCA); therefore, the new RCP shutdown seals are the primary reason for the significant reduction in MUCDF for these sequences (approximately 94 percent).

For seismic events, FLEX is most effective for the lower bins (seismic bins 1–4), where the reduction in MUCDF is in the 24–30 percent range. At the elevated seismic bins, a greater contribution of MUCDF comes from seismic failures that do not benefit from the modifications associated with the FLEX sensitivity case (in fact, there is little or no impact from FLEX on seismic bins 6–8).

Table 6-8 provides additional information regarding the FLEX MUCDF results for seismic events only. First, for each seismic bin, the contribution to FLEX MUCDF is provided for the different types of MU dependencies (e.g., structural failures, cross-unit CCFs, and HFEs). Some observations from this information include:

- Structural failures contribute to all (or almost all) of the MUCDF for seismic bins 7 and 8.
- Structural failures have no (or almost no) contribution to MUCDF for seismic bins 1 and 2.
- MU CCFs make some contribution to the MUCDF for seismic bins 1 through 3 but make no (or almost no) contribution to MUCDF for seismic bins 4 through 8.
- Cross-unit human dependencies make no contribution to seismic bins 4 through 8 and very little contribution to seismic bins 1 through 3.

For seismic bin 2 MUCDF results only, Table 6-8 shows the contributions from these same dependencies for both the base and FLEX sensitivity cases. For all MU dependencies explored, the percent contribution is essentially the same (e.g., the “seismic” contribution is around 60 percent and cross-unit CCFs contribute around 30 percent).

Table 6-7 MUCDF Estimates – Base Case and FLEX Case

			Base Case			FLEX Case	FLEX Effectiveness	Base Case CCDP
Scenario Name	Scenario Description	MU Scenario Characteristics	MUIEF (/rcy) f	MUCDF (/rcy) a	% Total MUCDF c	MUCDF (/rcy) b	(a-b)/a	a/f
MU-IE-LOOPGR	Grid-related LOOP	SBO and AC power recovery failure	6.15E-03	1.00E-06	7.7%	1.45E-07	85.5%	1.63E-04
MU-IE-LOOPPC	Plant-centered LOOP	SBO and AC power recovery failure	1.07E-04	1.43E-08	0.1%	1.42E-09	90.1%	1.33E-04
MU-IE-LOOPSC	Switchyard-centered LOOP	SBO and AC power recovery failure	2.80E-03	3.57E-07	2.7%	4.81E-08	86.5%	1.27E-04
MU-IE-LOOPWR	Weather-related LOOP	SBO and AC power recovery failure	2.44E-03	4.47E-07	3.4%	5.65E-08	87.4%	1.83E-04
MU-LONSCW	Loss of NSCW	Loss of NSCW in both units	3.47E-05	3.23E-06	24.8%	1.87E-07	94.2%	9.30E-02
MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned (CCDP=1)	1.47E-07	1.47E-07	1.1%			1.00E+00
MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2	at least MU LOOP (assumed)	3.42E-02	2.28E-08	0.2%			6.67E-07
MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least other unit reactor trip and fire damage (assumed)	9.08E-03	6.59E-08	0.5%			7.26E-06
MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least other unit reactor trip and fire damage (assumed)	9.08E-03	6.59E-08	0.5%			7.26E-06
MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic bin 1	1.64E-03	8.03E-08	0.6%	5.60E-08	30.3%	4.93E-05
MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic bin 2	2.19E-04	1.24E-07	1.0%	9.35E-08	24.3%	5.64E-04

Table 6-7 MUCDF Estimates – Base Case and FLEX Case (cont.)

			Base Case			FLEX Case	FLEX Effectiveness	Base Case CCDP
Scenario Name	Scenario Description	MU Scenario Characteristics	MUIEF (/rcy) f	MUCDF (/rcy) a	% Total MUCDF c	MUCDF (/rcy) b	(a-b)/a	a/f
MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic bin 3	4.79E-05	8.24E-07	6.3%	5.91E-07	28.5%	1.72E-02
MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic bin 4	1.34E-05	1.85E-06	14.2%	1.36E-06	26.5%	1.38E-01
MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic bin 5	4.26E-06	2.02E-06	15.6%	1.70E-06	15.8%	4.75E-01
MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic bin 6	1.9E-06	1.72E-06	13.3%	1.68E-06	2.6%	8.97E-01
MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2-unit SBO and major structural damage in seismic bin 7 (CCDP=1)	2.48E-07	2.34E-07	1.8%	2.34E-07	0.0%	9.43E-01
MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and major structural damage in seismic bin 8 (CCDP=1)	2.32E-09	2.32E-09	<0.1%	2.32E-09	0.0%	1.00E+00
MU-IE-WIND-1	SBO and SSC wind damage	SBO and wind damage to SSCs	8.89E-03	7.93E-07	6.1%	2.32E-07	70.7%	8.92E-05
		Total =	**7.47E-02**	**1.30E-05**	**100.0%**	**6.53E-06**	**49.7%**	**1.74E-04**

Table 6-8 Observations for the MU FLEX Seismic Scenarios

		BIN-1	BIN-2	BIN-3	BIN-4	BIN-5	BIN-6	BIN-7	BIN-8
pga =		**0.17g**	**0.39g**	**0.59g**	**0.79g**	**1.0g**	**1.29g**	**1.94g**	**2.5+g**
CDF due to "STRUCTURE" failures (direct CD) (/rcy)		0.00E+00	0.00E+00	8.08E-08	2.74E-07	3.76E-07	7.21E-07	2.34E-07	2.32E-09
Bin CCDP for structural failures		0.00E+00	0.00E+00	0.002	0.02	0.09	0.38	0.94	1.00
"STRUCTURE" contribution		none	ignorable	some	some	some	some	almost ALL	ALL
"SEISMIC" contribution		NO	YES	YES	YES	YES	YES		
"MU CCF" contribution		some	some	some	very little	none	none	none	none
"HEP" contribution		almost none	almost none	almost none	almost none	none	none	none	none
Seismically-induced LOOP probability		0.13	0.70	0.92	0.98	1.00	1.00	1.00	1.00
Bin-2 contributors (only)			**WITH-FLEX**	**NO-FLEX**					
		Seismic	64%	60%					
		CCF	29%	31%					
		Others	2%	5%					
		Structure	5%	4%					
		Total	100%	100%					
		Total MUCDF (/rcy)	9.35E-08	1.24E-07					
		FLEX Effectiveness	24%						

7 IDENTIFYING AN ILLUSTRATIVE MULTI-SOURCE SCENARIO FOR COMBINING MULTI-UNIT AND SPENT FUEL POOL RISK

This section describes the fifth step in the overall ISR task, providing a description of the approach used to integrate MU Level 1 and 2 risk results with the Level 1 and Level 2 SFP risk results. In particular, the L3PRA project's ISR task is focused on scenarios in which all radiological sources are challenged (more or less) simultaneously. In addition, since no sitewide dependencies (except for common IEs) between the dry cask storage (DCS) facility and the two reactors were found, the DCS facility has not been included in the illustrative multi-source scenario.

This step sets the stage for later steps, including the final step in the ISR task (i.e., calculation of multi-source risk). In particular, this step consists of the selection and definition of an illustrative scenario that involves both reactors and the SFPs. This step is important in limiting the scope of Level 2 and Level 3 PRA calculations in later steps to be consistent with project resources and computational abilities.

Results of previous ISR tasks and prior L3PRA project PRAs were used to perform this step, including the sitewide dependencies identified in Section 4 and preliminary MU Level 2 results. Examples of prior L3PRA PRA reports used include the single-unit external hazards Level 2 PRA report (NRC, 2023c), the report for the Level 1 and Level 2 spent fuel pool PRA (NRC, 2025a), and the dry cask storage PRA (NRC, 2024).

Section 7.1 provides some background for this step; specifically, information on two related efforts. Section 7.2 summarizes the approach for selecting and defining an illustrative multi-source scenario to demonstrate the integration of MU and SFP risk. Section 7.3 documents some scope limitations for the ISR task that greatly influenced the selection and definition of the illustrative multi-source scenario. Section 7.4 describes the illustrative scenario, while Section 7.5 provides the basis for certain elements of the scenario.

7.1 Background

There is limited experience in integrating reactor risk with SFP risk. This section briefly discusses two such efforts that integrate a single-unit Level 2 PRA model with an SFP PRA:

1. EPRI, "PWR Spent Fuel Pool Risk Assessment Integration Framework and Pilot Plant" (EPRI, 2014)

2. Basic, et al., "Assessment of potential impact of combustible gases from reactor core damage on risk of outside containment spent fuel pool damage" (Basic, 2023)

7.1.1 EPRI's 2014 Report on PWR Spent Fuel Pool Risk Assessment

The stated purpose of EPRI (2014) is "…to develop a generic framework and methodology for conducting an SFP-reactor PRA for pressurized water reactors (PWRs)" in recognition of the potential dependencies between the SFPs and reactors." EPRI (2014) also references a prior EPRI publication (EPRI, 2013) that addresses how to conduct an SFP-reactor PRA for boiling water reactors (BWRs).

EPRI (2014) is referenced by the L3PRA project's Level 1 and Level 2 SFP PRA report (NRC, 2025a) as it also provides guidance on how to perform a single source SFP PRA. Example topics for performing a single source SFP PRA that are provided in EPRI (2014) are:

- identification of initiating events (IEs) that can result in fuel damage in the SFP and their frequencies
- consequences of identified IEs (e.g., spent fuel damage)
- success criteria needed to model SFP risk
- mitigation measures used to prevent spent fuel damage

Regarding the combined SFP-reactor PRA modeling, EPRI (2014) provides guidance on identifying common IEs and other dependencies between a reactor and an SFP. Some of this guidance is similar to that used by the ISR task (as documented in Section 4 of the ISR report). In brief, EPRI (2014) links the end states of the Level 2 PRA's containment event tree with a simple event tree for the SFP. Dependencies between the reactor and the SFP are used to preemptively fail either of the two SFP mitigation strategies: SFP makeup or cooling.

Overall, the scope addressed by the guidance given in EPRI (2014) is much broader than that addressed by the L3PRA project's ISR task. For example, a wider range of IEs and operational states are addressed by the guidance given in EPRI (2014). However, the pilot application provided by EPRI (2014) addressed only at-power conditions (similar to the scope of the ISR task). Also, EPRI (2014) addressed only a single reactor and SFP combination. The L3PRA project's ISR task addresses both reactors on the reference site coupled with the two, hydraulically connected SFPs.[39]

Quantitative results for internal events and seismic events are given in EPRI (2014), including the following breakdown of results: (1) total CDF from all causes for the single reactor, (2) total fuel damage frequency (FDF) from all causes for the single SFP, (3) FDF considering causes that only affect the SFP, (4) CDF considering causes that only affect the reactor, (5) events that simultaneously affect the reactor and SFP, (6) CDF without SFP damage, and (7) incremental FDF due to serious accident affects from the reactor. Some of the key insights from the at-power, SFP-reactor, PWR application in EPRI (2014) are:

- Overall, as shown in Table 6-2 and described in Section 6.7.1 of EPRI (2014):
    - The risk results for the SFP are dominated by seismic events.[40]
    - The risk results for the reactor are approximately the same for internal events and seismic events, with the contribution from seismic events slightly higher.

[39] Note that EPRI (2014) calculated fuel damage frequency, rather than the metric of significant fuel uncovery frequency used in the L3PRA project for the SFP PRA.

[40] Note that the seismic calculations performed for EPRI (2014) are based on an older seismic model that uses five seismic initiator bins (as compared to the newer model used for the L3PRA project and its associated eight bins, as well as associated updated methodology and data).

- The risk results show that there is no impact from the SFP on the reactor for internal events, and very little impact from the SFP on the reactor for seismic events.

- On the other hand, the impact of the reactor accident progression on the SFP is about 16 percent for seismic events and about 35 percent for internal events.

- For only the SFP risk results (i.e., fuel damage frequency):

  - Seismic events (all five bins): Fuel damage frequency is termed “low, but non-negligible ($9.18{\times}10^{-7}$/year)” in EPRI (2014). The contribution from a simultaneous, severe reactor accident progression to SFP fuel damage frequency (i.e., $1.46{\times}10^{-7}$/year) is also termed by EPRI (2014) to be “non-negligible,” representing about 16 percent of the total fuel damage frequency for seismic events.

  - Internal events: EPRI (2014) states that the total SFP fuel damage frequency is “low” (i.e., approximately $6{\times}10^{-10}$/year) and the impact of severe reactor accident progression on the SFP results is approximately 35 percent.

- For LERF contributors (see Section 6.7.4 in EPRI [2016]):

  - EPRI (2014) emphasizes that the “… SFP LERF is over an order of magnitude higher than the reactor LERF.”

  - The EPRI (2014) results for LERF are “… dominated by the seismic-induced SFP LERF results (i.e., approximately 96% of the total LERF).”

EPRI (2014) provides other useful information, such as a discussion of modeling uncertainties and results of sensitivity studies.

7.1.2 2023 Paper on Potential Impact of Combustible Gases from Reactor on SFPs

Basic (2023) uses the general approach in EPRI (2014) (discussed above) to investigate specific potential dependencies between a reactor and an SFP. In particular, the stated objective of Basic (2023) is to “…[assess] the possible impact of a severe accident in the reactor, which is in the containment, to SFP outside the containment” (typically in the fuel handling building [FHB] for Westinghouse PWRs). A particular concern that was assessed is the possible leakage of combustible gases (such as hydrogen or carbon monoxide) from the containment of a damaged reactor. Such leakages could impact SFP integrity and/or the ability to perform mitigating actions. Basic (2023) notes that, while the fuel damage frequency (FDF) for SFPs for most IEs is low, it can be comparable to that for reactors if the IE is a seismic event.

Basic (2023) also used the approach in EPRI (2014) to identify what the L3PRA project’s ISR task defines as “human and organizational” potential sitewide dependencies. In particular, Basic (2023) states the following:

> Various post Fukushima Daiichi NPP accident studies (e.g., Buongiorno [2011]; INPO [2011]; ENSREG [2012]) concluded that plant TSC in an extreme and rare external event design-extension condition (DEC) could have a problem with the management of

coincident loss of decay heat removal from the core (possibly resulting in significant core damage) and a loss of decay heat removal from the spent fuel pool. From the point of view of prioritizing severe accident management strategies, the priority mitigation action should be to re-establish the emergency core cooling in the reactor pressure vessel. The apparent reason is the considerable time window available before the water inventory in the spent fuel pool would be evaporated and spent fuel exposed to overheating.

The analysis reported in this paper includes grouping reactor/containment event tree sequences (for a single reactor unit) with SFP states. For the analysis in this paper, the SFP event tree (e.g., modeling of SFP cooling and makeup functions) was modified for the reactor sequences that represented containment failures and the potential for combustible gases.

Overall, Basic (2023) used the approach in EPRI (2014) to investigate one potential dependency between the SFPs and reactors that is similar to, but not the same as, those sitewide dependencies that the L3PRA project's ISR task identified and addressed.

7.2 Approach

The approach used to select and define an illustrative multi-source scenario that allows for the integration of MU and SFP Level 1 and Level 2 risk relies on previous work performed, such as:

- the sitewide dependency assessment discussed in Section 4
- the identification of MU and sitewide IEs discussed in Section 5
- the MU Level 1 risk results reported in Section 6
- the details of the single source SFP Level 1 and 2 PRA that are documented in NRC (2025a)

The goal of the approach was to identify and describe scenarios for which both reactors and the SFPs could have radiological releases. The sitewide dependency assessment performed for the ISR task accomplished much of this work. Details from the SFP Level 1 and 2 PRA and the single reactor Level 2 PRAs were used to identify additional contextual details. The task leads for the ISR task, MU Level 2 PRAs, and SFP Level 1 and PRA held several brainstorming sessions to assist in developing a scenario that merges details for the accident progression for both the reactors and the SFPs.

While no PRA logic models were developed to integrate the MU and SFP Level 1 and Level 2 results, the process used to develop the illustrative multi-source scenario is consistent with the approach in EPRI (2014). In particular, MU Level 1 and Level 2 scenario details were reviewed to identify contexts that could fail SFP mitigation strategies (which is similar to EPRI's connection between single reactor containment event trees and the SFP event tree). The simplicity of the SFP event trees and the lack of certain types of dependencies between the reference plant's reactors and SFPs allows use of a simpler, less formal approach for the purposes of the L3PRA project's ISR task. Note, however, that EPRI (2014) was used only for one single reactor-single SFP combination. The ISR task, in contrast, addresses simultaneous failures of both reactors and both (hydraulically connected) SFPs.

The illustrative multi-source scenario used to develop integrated MU and SFP Level 3 PRA results is described in Section 7.4. The next section provides further information on scope limitations for the ISR task that contributed to the definition of the illustrative multi-source scenario.

7.3 Scope Limitations for the Multi-Source Scenario

Section 3.1 describes the overall scope of the ISR task for the L3PRA project. However, there are specific scope items given in Section 3.1 that are important to reiterate when considering inputs from the SFPs for calculation of multi-source risk. Also, there are additional scope choices that are relevant specifically to the MU Level 2 PRA effort, as well as to later ISR tasks.

In particular, two key scope limitations specific to the overall ISR task are:

- The ISR task addresses only at-power conditions for both the reactors and the SFPs since the L3PRA project reactor, low-power and shutdown PRA (NRC, 2025c) does not address seismic events, which are the major contributors to SFP risk.

- The ISR task does not address combinations of only one reactor unit with either the SFPs or dry cask storage.[41]

In addition, the L3PRA project did not perform a FLEX sensitivity study for the SFPs. However, as indicated by discussions provided elsewhere in the report (e.g., Section 4.5, Section 4.6, Section 12D.2 , and Section 12F.4), the project team reviewed the licensee's publicly available FIP, including aspects related to makeup to the SFPs. In particular, the licensee's FIP identifies more water sources for providing makeup to the SFPs than are specified in the EDMGs. Based on this information, the project team expects that implementation of FLEX strategies would eliminate some potential dependencies between the reactors and the SFPs with respect to resource sharing.

There are important implications regarding the first scope limitation identified above. Namely, as shown in Table 3-1, the scope only includes operating cycle phases (OCPs) defined as "nominal" in the SFP Level 1 and Level 2 PRA report (i.e., not when either reactor is shutdown or when a cask is being loaded). Table 7-1 below duplicates Table 3-2 in the SFP Level 1 and Level 2 PRA report (NRC, 2025a) which shows the different OCPs that involve the different operating states of the two reactors and two spent fuel pools on the reference site. Highlighting has been added to Table 7-1 showing that there are several different OCPs for which both reactors are at power and both SFPs are in a "nominal" state, namely:

- AAN1
- AAN2
- AAN3
- AAN4
- AAN5

The scope limitation to at-power only conditions for the two reactors is especially notable because, as shown by the SFP results in NRC (2025a), the shutdown phases contribute

[41] As will be seen in the discussion below, this scope limitation does not end up being a real constraint (i.e., scenarios that would involve both the SFPs and the reactors generally would be so severe that it is likely that both reactors would experience plant damage rather than only one).

significantly to SFP risk (e.g., shutdown phases have higher frequencies of significant fuel uncovery and have associated larger releases on average), as compared to other phases.

Table 7-1 Operating Cycle Phase Discretization

<table>
<tr><th>Site Phase Identifier[1]</th><th>Timeframe (days)</th><th>Unit 1 Reactor</th><th>Unit 2 Reactor</th><th>Unit 1 SFP</th><th>Unit 2 SFP</th><th>DCS</th></tr>
<tr><td>SAO</td><td>0 – 6</td><td rowspan="3">Shutdown</td><td rowspan="5">At-power</td><td colspan="2">Outage entry (U1)</td><td rowspan="10">Storage</td></tr>
<tr><td>SAR1/SAR2[2]</td><td>6 – 16</td><td colspan="2">Refueling (U1)</td></tr>
<tr><td>SAP</td><td>16 – 30</td><td colspan="2">Post-refueling (U1)</td></tr>
<tr><td>AAN1</td><td>30 – 80</td><td rowspan="9">At-power</td><td colspan="2" rowspan="2">Nominal</td></tr>
<tr><td>AAN2</td><td>80 – 180</td></tr>
<tr><td>ASO</td><td>180 – 186</td><td rowspan="3">Shutdown</td><td colspan="2">Outage entry (U2)</td></tr>
<tr><td>ASR1/ASR2[2]</td><td>186 – 196</td><td colspan="2">Refueling (U2)</td></tr>
<tr><td>ASP</td><td>196 – 210</td><td colspan="2">Post-refueling (U2)</td></tr>
<tr><td>AAN3</td><td>210 – 260</td><td rowspan="4">At-power</td><td colspan="2" rowspan="2">Nominal</td></tr>
<tr><td>AAN4</td><td>260 – 360</td></tr>
<tr><td>AAC</td><td>360 – 400</td><td colspan="3">Cask Loading</td></tr>
<tr><td>AAN5</td><td>400 – 548</td><td colspan="2">Nominal</td><td>Storage</td></tr>
</table>

[1] The site phase identifiers are defined as follows:

Character 1: Unit 1 is shutdown (S) or at power (A).
Character 2: Unit 2 is shutdown (S) or at power (A).
Character 3: The shutdown unit has entered the outage (O), is in refueling (R), or is in post-refueling (P). If neither unit is in shutdown, cask loading is occurring (C) or is not occurring (N).
Character 4: Distinguishes different timeframes for phase AAN.

[2] OCPs SAR2 and ASR2 are the same as SAR1 and ASR1, except the fuel transfer tube for the refueling reactor is closed in OCP SAR2 and ASR2. The rest of the analysis assumes the fuel transfer tube is open for the entirety of SAR and ASR. Accordingly, the analysis assumes the SFPs may be affected by events in the reactor and uses SAR1 and ASR1 timings which are somewhat longer as more water is available. In reality, the tube will be open during defueling and refueling but may be closed in between.

7.4 Illustrative Multi-Source Scenario

Section 7.4.1 describes the multi-source scenario selected to be addressed in the section on multi-source risk integration (i.e., Section 9). Section 7.4.2 provides a brief discussion of some other scenarios that would be relevant to address if project resources were not limited.

7.4.1 Description of Illustrative Multi-Source Scenario

The scenario that was selected and developed for the ISR task is summarized in Table 7-2. Table 7-2 consists of a timeline of major events, especially those that are important to modeling inputs used in Level 1, Level 2, and Level 3 PRAs for both the reactors and the SFPs. Section 7.5 provides a discussion on why certain choices were made in developing the illustrative multi-source scenario.

The accident progression for the reactors and the SFPs shown in Table 7-2 is set up by the initial conditions (i.e., both reactors at power and SFPs in nominal conditions), the specific seismic event (i.e., seismic bin 6), and response to the seismic event. For simplicity, both reactors are assumed to experience core damage and containment failure at the same time.

Regarding the initial conditions, the choice of “at-power” for the reactors and “nominal” for the SFPs was previously explained in the discussion of project scope above. However, there are important implications that have been discussed in other report sections (e.g., Section 7.3). In short, at-power and nominal conditions is a “safer” state with regard to release paths and potential dependencies between the reactors and the SFPs. For these reasons (among others), consideration of shutdown conditions for MU and MU-SFP PRAs is a recommended candidate for future work.

The selection of seismic bin 6 for the illustrative multi-source scenario, as explained in detail in Section 7.5, prescribed many of the needed event details for quantifying the joint risk of both reactors and the SFPs. The most important of those event details is occurrence of a large seismic event that:

- is a significant contributor to MU and SFP risk
- causes widespread site structure, system, and component (SSC) damage, including:
    - widespread SSC damage that limits the effectiveness of reactor Level 1 mitigation strategies
    - large SFP inventory losses that require mitigation
    - Loss of offsite power (LOOP) and station blackout (SBO) conditions which, in turn, limit the effective mitigation strategies for the SFPs, but do allow for the possibility of mitigation strategies being effective (i.e., a large, but not the largest, seismic event modeled)

The goal of the project team in selecting the illustrative multi-source scenario was to demonstrate how potential dependencies between the reactors and the SFPs could be reflected in integrated risk calculations. Based on the results of the sitewide dependency assessment given in Section 4, the project team focused on the sharing of physical resources and SSCs, and the associated sharing of human resources. In particular, under certain conditions (that match the characteristics of seismic bin 6), mitigation strategies for the reactors that are modeled in the Level 2 PRA involve the use of EDMG strategies. Those reactor EDMG strategies require use of the same equipment, water resources, and operators that are needed for the SFP EDMG mitigation strategy for makeup. In addition, as discussed in more detail below, required timing for both the reactor and SFP mitigation strategies is similar only for seismic bin 6. The discussion below also indicates how the reactor-SFP conditions and timing of mitigation strategies are intertwined.

Some of the reactor-related events shown in Table 7-2 have different timing for different release categories (RCs) (e.g., Containment Isolation Failure (CIF), Late Containment Failure (LCF), Intermediate Containment Failure due to Burn (ICF-BURN)). These release categories will be discussed further in Section 8. Section 9 uses release category frequencies for both the reactors and the SFPs to calculate multi-source consequences.

Note that EPRI’s report on the integration of single reactor Level 1 and Level 2 PRAs with SFP Level 1 and Level 2 PRAs (EPRI, 2014) also selected a seismic event for its pilot application.

Table 7-2 Timing of Key Events in the Illustrative Multi-Source Scenario

Time	Event(s)	Plant/SFP state	Notes
-0	Initial conditions	Both reactors at power	
		Both SFPs in "nominal" state (i.e., no fuel handling activities)	
0	**Large seismic event**	Seismic bin 6	
	Reactor trip	Both reactors (simultaneous)	
	Loss of offsite power	Sitewide event	These losses are relevant to both reactors and the SFPs.
	Station Blackout	Sitewide event	
	Significant water sloshing	Both SFPs	These failures are assumed to be simultaneous for the hydraulically-connected SFPs.
	Liner failures (e.g., leaks)	Both SFPs	
	Extensive structural damage from seismic event	Sitewide	Buildings relevant to both reactors and SFPs are affected.
	Extensive equipment failure from seismic event	Sitewide	Equipment relevant to both reactors and SFPs are affected.
1 hour	Technical Support Center is staffed		
3 hours	**General Emergency is declared**		
See "Event"	**Core damage at:** - CIF = 16 hours - LCF = 3.9 hours - ICF-BURN = 16 hours	Reactors	Both reactors are assumed to experience core damage at the same time for the same release categories (RCs). Core damage occurs later for CIF and ICF-BURN because, for these RCs, AFW is assumed to operate for 4 hours before batteries are depleted."

Table 7-2 Timing of Key Events in the Illustrative Multi-Source Scenario (cont.)

Time	Event(s)	Plant/SFP state	Notes
12 ½ hours	**Evacuation complete:** - 45 minutes assumed for notifications - Additional 8 ¾ hours to evacuate the 10 mile Emergency Planning Zone (EPZ)	SFPs and reactors	The same evacuation model is used for all seismic bins (which is referred to as the "degraded evacuation model").
~10 hours	**Time when external makeup for SFPs is required**	SFPs	This is an EDMG strategy that uses a B.5.b pump, FWSTs, etc.
~22 hours	**Time when containment spray cooling is required for reactors (for LCF and ICF-BURN only)**	Reactors	This is an EDMG strategy that uses a B.5.b pump, FWSTs, etc.
See "Event"	**Containment failure**: - CIF = 0 hours - LCF = 48 hours - ICF-BURN = 28 hours	Reactors	Both reactors are assumed to experience containment failure at the same time for the same RCs.
See "Event"	**Containment releases**: - CIF = 21 & 18 hours - LCF = 55 & 158 hours - ICF-BURN = 28 & 33 hours	Reactors Xenon and iodine releases, respectively, for each release category	Both reactors are assumed to experience containment releases at the same time for the same RCs.
N/A	Time when external spray strategy for SFPs is required	SFPs	This is an EDMG strategy that uses a B.5.b pump, FWSTs, etc. However, due to insufficient time, this strategy is failed for seismic bin 6.
See "Event"	**SFP releases:** - 60 hours (AAN1) - 153 hours (AAN5)	SFPs	Data taken from Table 3-32 in NRC (2025a).

7.4.2 Other Scenarios Important to Multi-Source Risk

Although project resources allowed only one illustrative multi-source scenario to be developed in assessing multi-source risk, other scenarios were identified that could merit development if resources allowed. Examples of such scenarios include:

- Seismic bin 5 for both reactors at power and SFPs in nominal conditions
- Seismic bins 7 and 8 for both reactors at power and SFPs in nominal conditions
- Shutdown scenarios for reactors and SFPs

An multi-source scenario for seismic bin 5 would be similar to that for seismic bin 6 except that the required timing for mitigation strategies would be longer. Based on the discussion in Section 7.5, mitigative actions for the SFPs would be required many hours later than for the reactors. However, as pointed out in Basic (2023) and illustrated by the Fukushima Daiichi event (e.g., INPO, 2011 and INPO, 2012), such differences in required timing do not preclude potential human and organizational dependencies between operator actions needed for the reactors and those needed for the SFPs.

Regarding seismic bins 7 and 8, they are predicted to be so large and destructive that their MU conditional core damage probability is 1.0 (see, for example Table 6-4). These large seismic events also have a devastating impact on the SFPs (despite their robust construction). As a result, the ability to take mitigative measures in the required timing, especially for the SFPs, is not credited in the SFP Level 1 and Level 2 PRA (NRC, 2025a). Consequently, the sequence of events for these scenarios would be similar to that given above for seismic bin 6 except that:

- damage is more severe and more likely
- timing of certain events is likely to be quicker
- no credit is given for mitigation

Because of the above, dependencies (beyond the common initiator) between reactors and the SFPs are irrelevant for seismic bins 7 and 8. Without such dependencies, integration of MU risk and SFP risk should be easier than for cases when there are dependencies that should be represented (e.g., seismic bin 6).

Finally, as noted in Section 7.3, omission of OCPs for shutdown states also omits a major portion of the calculated SFP risk. Treatment of multi-source risk for shutdown conditions is captured as a candidate for future work in Section 11.7.

7.5 Basis for Selection of Illustrative Multi-Source Scenario Elements

Due to limited resources, the ISR task addressed only one representative scenario that affects both reactors and the SFPs. In particular, one objective of the ISR task regarding the SFPs was to identify potential dependencies between the SFPs and the reactors that could change the results that have already been developed in the single source, SFP Level 1, Level 2, and Level 3 PRAs when considering multi-source risk.

As presented earlier in Section 4, a formal, systematic sitewide dependency assessment was performed for all radiological sources on the reference site (i.e., two reactors, two spent fuel pools, dry cask storage facility), for all hazards, and for Level 1 and Level 2 PRAs. This assessment was performed in support of the ISR task, generally, and, specifically, to identify potential dependencies between the two reactors, between the SFPs and the two reactors, and between the DCS and the two reactors.

The sections below describe the various inputs (and combinations of inputs) used to select and define the multi-source scenario, such as:

- results from the sitewide dependency assessment (Phases 1, 2 and 3)

- information from the SFP Level 1 and Level 2 PRA (NRC, 2025a), including HRA details such as required timing for mitigative measures

- information from the single source, reactor Level 1 and 2 PRAs, including timing of key events

- MU Level 1 and 2 risk results

- brainstorming meetings with leads for single source reactor and multi-unit Level 2 PRAs, SFP Level 1 and Level 2 PRAs, and reactor and SFP Level 3 PRAs

7.5.1 Focus on Seismic Events

All the potential multi-unit IEs (MUIEs) that were identified in the Phase 1 sitewide dependency assessment were used to develop multi-unit core damage frequencies (MUCDFs) (as shown in Table 6-3 of Section 6 of this report). However, as discussed in Section 2, resource and other limitations required a more limited number of MUIEs to be addressed in the development of multi-unit release category frequencies (MURCFs) (i.e., MU Level 2 risk) and, consequently, for MU Level 3 risk results. Also, the project team recognized that the scope of MUIEs addressed by Level 2 and Level 3 PRA also needed to include IEs that were important to SFP risk.

Consequently, important risk contributors within MUCDF and (single source) SFP releases were used to inform selection of the IEs addressed in MUCDF calculations (Section 6) and those that will be addressed in the final ISR task (Section 10) when MU and SFP risk are integrated.

First, the discussion in Section 6.3.1 identified the following main contributors to MUCDF results:

- The combination of all four LOOPs contributes about 14 percent to total MUCDF.
- The combination of all eight seismic bins contributes about 53 percent to total MUCDF.
- Loss of NSCW events contribute about 25 percent to the total MUCDF.

From these results, LOOPs and seismic events were identified as candidates for treatment by MU Level 2 and Level 3 PRA. Weather-related LOOPs (LOOPWRs) were selected as representative of LOOPs, in general.

These insights from the MUCDF results were then combined with the Phase 1 sitewide dependency assessment results for the two reactors, SFPs, and DCS that are summarized in Table 4-3. Table 4-3 is based on results presented in Appendix C, including Table C-4 and Table C-5, which document the results of the Phase 1 sitewide dependency assessment for the SFPs and DCS, respectively. Note that Table C-5 shows that the only MUIEs that have risk significance for the DCS facility are seismic events when fuel handling activities are occurring. However, as stated above, OCPs that involve cask handling are out-of-scope for the ISR task. Consequently, only risk contributions from the SFPs were addressed in the ISR task.

Table 7-3 is adapted from the Phase 1 sitewide dependency assessment results for SFPs given in Table C-4, which, in turn, is based on Table 3-34 and other information documented in the SFP Level 1 and Level 2 PRA report (NRC, 2025a). In particular, Table 7-3 shows that most of the SFP significant fuel uncovery frequency (SFUF) comes from seismic bins 5, 6, and 7, which together account for almost 94 percent of SFUF.

7.5.2 Why Seismic Event Bins 5 and 6 (and not 7)

As can be seen from the Table 7-3, the largest contributors to SFP SFUF are seismic bins 6 and 7. The next largest contributor is seismic bin 5. The remainder of the seismic events consist of seismic bin 8, which has an IE frequency too low to significantly contribute, and lower seismic bins (4, 3, and 1 and 2 combined). The relatively minor contribution of the lower seismic bins is not surprising, given that they are less likely to result in losses of inventory (see Table 3-21 of the SFP Level 1 and 2 PRA report [NRC, 2025a]) and the 7-day sequence truncation time eliminates scenarios that do not either leak or have a large inventory loss from sloshing.

As stated in Section 7.4.1, the illustrative multi-source scenario was selected based on how potential dependencies between the reactors and the SFPs could be reflected in integrated risk calculations, especially potential dependencies between the reactors and the SFPs with respect to the mitigative strategies required for accident response. This objective determined what type of information from the reactor and SFP PRAs needed to be considered to further narrow the scope of seismic events to consider.

Regarding mitigation strategies for the SFPs, Appendix G of NRC (2025a) documents the details of the HRA performed for the Level 1 and Level 2 SFP PRAs. Using the terminology used in the reference plant's EDMGs, the SFP HRA models operator actions related to two types of EDMG strategies: (1) the "internal strategy" (i.e., equipment predominantly located in the vicinity of the refuel floor in combination with installed systems) and (2) the "external strategy" (i.e., use of on-site portable equipment and installed tanks that are deliberately remote from the refuel floor).

However, the HRA for the SFPs did not credit mitigation strategies for seismic bins 7 and 8 (i.e., the most challenging seismic bins), based on the following: (1) the extreme sloshing in these cases will make the refuel floor immediately uninhabitable due to high radiation levels associated with low SFP water level (i.e., fuel uncovery is immediate), (2) the fuel handling building (FHB) itself may experience significant damage and thus be difficult to access, and (3) the potential for these extreme events to further degrade human performance. In other words, no operator actions are assumed to take place in the FHB when its temperature exceeds 125°F or when the level of either SFP is less than 4 feet above the fuel racks (i.e., when hazardous environmental conditions [HEC][42] are defined to occur).

[42] Hazardous environmental conditions are defined as fuel handling building air temperatures above 125°F or water level less than 4 feet above top of active fuel in spent fuel pools.

Table 7-3 Search for Potential Sitewide Initiating Events that Impact SFPs Using SFUF Results

L3PRA's MUIEs	Relevant to SFPs? (Y/N)	Refinement/ Caveat Notes	Contribution to Significant Fuel Uncovery Frequency (SFUF)	Other Notes
LOOP (grid-related only)	No. As discussed above these events were screened out of the SFP analysis.		Unknown but a sensitivity study in the SFP analysis suggests that the contribution would be small.	
LOOP (switchyard-related only)				
LOOP (weather-related only)				
Seismic events	Yes.	Bin 7	40.5%	
	Yes.	Bin 6	37.6%	
	Yes.	Bin 5	15.5%	
	Yes.	Bin 4	5.1%	
	Yes.	Bin 3	0.9%	Small contribution to SFUF
	Yes.	Bin 8	0.4%	Small contribution to SFUF
	Yes	Bins 1 and 2	0.0%	Negligible contribution to SFUF
Non-seismic LLOINV	Yes.		0.0%	Applicable during ASR/SAR (when shutdown unit is connected to the SFP).
Loss of NSCW	No.			Screened out by the 7-day truncation time.

Appendix A in the SFP Level 1 and Level 2 PRA report (NRC, 2025a) describes the approach and calculations used to predict the amount of sloshing in the SFPs for seismic events of different sizes. Table G-3 in NRC (2025a), duplicated here as Table 7-4 below, shows the SFP level with respect to the HEC definition for different seismic bins, as well as the associated sloshing heights.[43] Note that significant sloshing is predicted for only the larger seismic bins (e.g., bins 6, 7 and 8) which represent very large seismic events. In particular, for seismic bins 7 and 8, the conditional fuel damage probability (CFDP) is assumed to be 1.0 due to the amount of sloshing. To be clear, seismic bins 7 and 8 are important to the calculation of multi-source risk. However, for the purposes of the L3PRA project's ISR task, seismic bins 7 and 8 cannot provide insights on the impact of potential dependencies between the reactor and SFP mitigation strategies.

[43] Section 3.2.4 in the SFP Level 1 and Level 2 PRA report (NRC, 2025a) provides a general discussion of sloshing that can occur for the SFPs during very large seismic events (e.g., higher seismic bins). Appendix A of NRC (2025a) documents the sloshing calculations that were performed, resulting in wave heights of 33 feet for seismic bin 7 (as shown in Table 3-21 of that report).

While sloshing also is predicted for seismic bins 5 and 6, Table 7-4 shows it is estimated that the SFP level will remain 4 or 2 meters above the HEC definition for seismic bins 5 and 6, respectively. Consequently, mitigation strategies are possible for both of these seismic bins.

Table 7-4 Summary of Sloshing Estimates

	EQK 1[1]	EQK 2	EQK 3	EQK 4	EQK 5	EQK 6	EQK 7	EQK 8
Slosh height (ft)	1.0	2.3	3.2	6.7	9.4	15.6	33.0	n/a
Slosh height (m)	0.30	0.70	0.98	2.04	2.87	4.75	10.06	n/a
Water left (m)	11.7	11.3	11.1	10.0	9.2	7.3	2.0	n/a
Fraction of water left	0.975	0.942	0.919	0.830	0.762	0.605	0.165	n/a
Height above HEC (m)	6.6	6.2	6.0	4.9	4.1	2.2	-3.1	n/a
Height above SFU (m)	9.08	8.68	8.41	7.34	6.52	4.63	-0.68	n/a

[1] *EQK refers to earthquake bin.*

7.5.3 Seismic Bin 6

In order to choose the most appropriate seismic bin for the illustrative multi-source scenario, the project team needed to consider further details of the SFP mitigation strategies. For larger seismic events, such as seismic bins 5 and 6, LOOP is almost certain (i.e., about 94 percent of scenarios) and the probability of SBO is high (e.g., about 70 percent of scenarios) The project team decided to focus on those scenarios for which both a LOOP and SBO occurs. Since the “internal” mitigation strategies for the SFPs require power, they cannot be included in the accident response. Consequently, only “external” mitigation strategies for the SFPs, which use portable equipment, can be used in response to such seismic events.

The project team re-examined the potential dependencies identified between the reactors and the SFPs. Section 4.5.2 discusses the results of the Phase 2 sitewide dependencies (i.e., potential shared or connected SSCs), both between the two reactors and between the SFPs and reactors. In particular, the reactor Level 2 PRA mitigation strategy and SFP external strategies share water resources and the B.5.b pumps. For example, as noted in Section 4.6.2.1, there are only two B.5.b pumps and they may both be needed for the reactors. In addition, Section 4.6.2.1 discusses the potential for human and organizational dependencies between the reactors and between the reactors and the SFPs implied by these shared water sources and equipment (i.e., there also may be sharing of available operators).

For seismic bin 6, approximately 10 percent of SFP scenarios involve a liner failure that produces a leak (in addition to sloshing) that requires SFP inventory makeup. The project team decided to focus on this type of scenario because SFP inventory makeup via external strategies results in potential dependencies between the SFPs and reactors. For the selected scenario, the SFPs and reactors share water resources, equipment, and operators for accident response. In addition, these resources may not be adequate if needed for accident mitigation by all radiological sources on site. The SFP Level 1 and Level 2 PRAs did not include consideration of a multi-source accident in the estimation of the time needed to take mitigative actions. If there is a concurrent reactor accident (or two), operators will likely take longer to perform any actions for the SFP than reported in the SFP Level 1 and Level 2 PRAs. Also, because mitigation of reactor conditions was assumed to be the highest priority for the site, operator actions needed for the SFPs might not be performed at all if the only available operators are being used to address a reactor accident.[44] Consequently, if operator actions and associated SSCs for accident mitigation are needed for all three radiological sources (i.e., both reactors and the SFPs) in the same timeframe, the mitigation strategies for the SFPs would fail due to lack of resources (i.e., water, equipment, and operators).

The required timing for the EDMG containment spray strategy for the reactors shown in Table 7-2 is taken from the internal events and floods Level 2 PRA (NRC, 2022b). Regarding the EDMG containment spray strategy, MELCOR results were reviewed by the task lead who concluded that the appropriate time to credit containment spray using firewater is 22 hours after the initiating event. This time is shortly after vessel breach for all three of the multi-unit release categories used for the multi-source scenario (i.e., CIF, LCF, ICF-BURN). Success of the spray action moves a sequence into the corresponding scrubbed release category (i.e., CIF-SC, LCF-SC, ICF-BURN-SC). However, in the multi-unit results for seismic bin 6, those scrubbed release categories had much lower frequencies, suggesting that failure of the spray action is very likely regardless of competition with the SFP.

Based on the above, the project team considered the required timing of SFP mitigation strategies with respect to the timing of reactor mitigation strategies. The SFP Level 1 and 2 PRA report (NRC, 2025a) reported times by which operator actions must be completed, as calculated with a simplified MELCOR model. The amount of time available to align makeup or spray varied, depending on the size of the leak, the amount of sloshing (affected by the seismic bin), and the OCP (which affects decay heat). A series of tables in Appendix G of NRC (2025a) provide this timing information. For example, Table G-36 provides timing information for performing the external makeup strategy for a 200-gpm leak (in addition to the sloshing shown in Table 7-4 above) and shows the following ranges of times for the relevant OCPs for the ISR task:

- Seismic bin 5: 1112 – 1133 minutes (i.e., greater than 18 hours)
- Seismic bin 6: 568 – 613 minutes (i.e., approximately 10 hours)

Because the time available for seismic bin 6 is less than that for seismic bin 5 (and likely corresponds with the time when operator actions are needed for makeup strategies for the reactors), seismic bin 6 was selected by the project team as more likely to have dependencies between the SFPs and the reactors regarding the sharing of resources and staff.[45] In particular, Table 7-2 (above) shows that:

- Core damage occurs at either approximately 4 hours or approximately 16 hours, depending on the reactor release category.
- The reactors require mitigation strategies to be completed by about 22 hours after reactor trip.

[44] It should be noted that the human error probability developed for the EDMG strategies for seismic bin 6 is already very close to 1.0. Consequently, the overall failure probability for the SFP EDMG strategies cannot get much larger.

[45] From Section G.7.5 in NRC (2025a), there is insufficient time for the external spray strategy for seismic bin 6.

- The time by which the EDMG makeup strategy for the SFPs must be completed is approximately 10 hours after reactor trip.

Therefore, the need to implement the SFP EDMG strategy occurs before the need to implement the EDMG strategy for the reactors (and probably while operators are busy with actions related to the reactors).

Note, in the SFP report (NRC, 2025a), a simplified MELCOR model was run to calculate timings for a few different leak sizes and sloshing levels. The time to hazardous conditions is affected by the OCP of the pools (this affects the decay heat and the configuration, which affects the amount of water connected to the pools) and the seismic bin, which affects the amount of sloshing. The simplified MELCOR model was run for only a few OCPs and sloshing amounts—all the remaining combinations were calculated using a spreadsheet. This process is discussed in Section 3.4.2 of NRC (2025a). Appendix G of that report provides the results of these calculations and has tables for each scenario, showing the amount of time available to take actions, as well as the performance shaping factor that results from this time available.

One important note about the HRA for the SFP PRA is that the diagnosis human error probability (HEP) (which is approximately 0.5) is a lot higher than the execution HEP. The same diagnosis HEP was used for all HFEs modeled in the SFP PRA and is based on assuming the maximum amount of time for diagnosis. Therefore, the HEPs for all HFEs in the SFP PRA are quite high and are dominated by the diagnosis contribution.

7.6 Summary for Selection of Illustrative Multi-Source Scenario

The L3PRA project team identified a unique multi-source scenario that involves nearly simultaneous (e.g., within the traditional 24-hour mission time) consequences at both reactors and SFPs. Specifically, the multi-source scenario is for a seismic bin 6 event that:

- involves the sitewide dependencies identified in this section
- can be mitigated but requires mitigation for SFPs within 24 hours, unlike most of the accident scenarios addressed by the L3PRA SFP PRA, where mitigation is not required for many hours after accident initiation
- involves a timing dependency (i.e., given the amount of time it takes to complete the mitigative actions, they can be considered to occur in the similar timeframe):
    - the reactors require mitigation strategies to be completed by about 22 hours after reactor trip
    - the SFPs require mitigation strategies to be completed by about 10 hours after reactor trip
- involves releases from both reactors and the SFPs

8 MULTI-UNIT LEVEL 2 PRA

This section describes the sixth step in the overall ISR task, providing a description of the approach used to calculate MU Level 2 PRA risk and the associated results. Results of previous ISR tasks were used to develop MU Level 2 risk results, especially the MUCDF results provided in Section 6, MU and sitewide IE frequencies that were developed in Section 5, and other sitewide dependencies identified in Section 4. Appendix K provides supporting details for this section.

8.1 Introduction

This portion of the multi-source risk assessment has four primary objectives:

- Evaluate the potential for cross-unit dependencies arising from severe accidents or affecting severe accident progression.
- Select single unit release categories (RCs) to develop multi-unit RCs for illustrative calculations.
- Demonstrate a method for quantification of multi-unit release category frequencies (MURCFs).
- Provide, in conjunction with the multi-unit Level 3 results given in Section 9, some insight into the risk significance of multi-unit releases in terms of offsite consequences.

In the current proof-of-concept study, quantification of MURCFs is limited to two, representative multi-unit initiating events:

1. weather-related LOOPs
2. an earthquake in seismic bin 6

The former is a significant contributor to MUCDF (along with other types of LOOPs), while the latter is a rare but highly consequential event that could plausibly affect many areas of the site, including both units and the SFPs. Further discussion on the selection of the seismic bin 6 initiating event was provided in Section 7.

The remainder of Section 8 is organized as follows:

- Section 8.2 examines cross-unit dependencies in severe accidents, including both causal dependencies and common cause failures (CCFs).
- Section 8.3 presents the quantification approach and results for MURCFs.
- Section 8.4 discusses the risk significance of the results, particularly in terms of large early release frequency (LERF) and large release frequency (LRF) implications.
- Section 8.5 addresses potential uncertainties in the analysis and possible sensitivity studies.

8.2 Cross-Unit Dependencies in Severe Accidents

Dependencies between units at a site can be divided into CCFs, in which similar system or component failures at the two units are caused by some third factor that affects both units symmetrically, and cascading failures (or causal dependencies), in which a failure at one unit causes a second, and not necessarily similar, later failure at the other unit (Zhou, 2021).

This section examines two major types of cross-unit dependencies. Section 8.2.1 explores causal dependencies, including depletion of shared resources, radiation hazards, and combustion effects. Section 8.2.2 discusses cross-unit basic event coupling due to common cause failures and external events, with specific focus on weather-related LOOP and seismic bin 6 events.

8.2.1 Causal Dependencies

Within a single unit PRA, causal dependencies are modeled primarily by fault tree or event tree logic, but taking the same approach for multiple units would likely result in excessive complexity of the PRA logic. For this discussion, “causal dependencies” would be addressed in either Phase 2 or Phase 3 sitewide dependency assessments. Examples of sitewide dependency categories that could be termed causal dependencies are cascading failures, proximity failures, and shared resources. While Section 4.6 provided such results, these types of sitewide dependencies have been re-examined or repeated in the context of Level 2 PRA.

For the MUCDF results given in Section 6, the only cascading types of failures addressed are those for certain fire scenarios (see the sitewide dependency assessment for cascading failures discussed in Section 12F.3.3.3). However, the Level 2 PRA analysis also should ideally include the potential for a core damage accident at one unit to cause cascading effects on the other unit, either as an initiating event (if the original initiating event was single-unit) or by compromising mitigating systems at the other unit after a multi-unit initiator. These cross-unit dependencies have not been extensively researched in the past (Zhou, 2021), but this section will explore some possible mechanisms and attempt to loosely estimate their risk significance.

The intention of the single unit PRA is to capture the CDF from all relevant initiating events and subsequent system failures at one unit, which would include those caused by (propagated from) the other unit. As stated in Section 4.6.3.2, the sitewide dependency assessment based on the Level 1 PRAs concluded that certain fire scenarios were the only cascading type of MUIE. But that assessment did not fully address potential dependencies for Level 2 PRA (e.g., it did not explicitly account for initiating events or other failures that are caused by *post-core damage phenomena* at the other unit). If these effects could be quantified, they would add not only to MUCDF but also to single unit CDF.

For single unit initiating events, the potential to create cascading failures at another unit is presumably bounded by the initiating event’s probability of causing both core damage and containment failure, since an accident progression that does not result in containment failure will have minimal physical effects on SSCs outside the containment building. Most releases to the environment also occur long after the initiating event, allowing time for mitigating actions at the second unit, and are dispersed in the environment so that they have limited impact on shared equipment. The rare exceptions may include containment isolation failure and interfacing systems LOCAs (ISLOCAs), as discussed in Section 8.2.1.3 on combustion given below. It can therefore be concluded that MUCDF due to cascading failures from a single unit initiating event is much smaller than the single unit LRF, and probably not a major contributor to total MUCDF because single unit ISLOCA and isolation failure frequencies are lower than MUCDF.

8.2.1.1 Depletion of Shared Resources

Another mechanism for sitewide dependencies concerns finite resources that are shared between units. In a multi-unit accident, such resources might be available for whichever unit needs them first, but not easily replaced before they are required at the other unit. Such dependencies were identified in Section 4. Examples of such resources are staffing and water.

Staffing: Section 3.5.5 of NRC (2023c) notes that the Level 2 HRA for external events did not account for staff injuries or staffing shortages, which can be a particular problem after major seismic or wind events that disrupt surrounding infrastructure. The problem of staff availability may be exacerbated in the case of a multi-unit core damage accident, since some of the same staff would be needed for mitigation efforts at both units.

Water sources: The only shared water source identified as a concern is the use of the B.5.b pump for containment spray from firewater storage tanks (where this strategy is defined with the basic event, 1-L2-OP-SCG1-1). The success criteria for this action includes use of both storage tanks, so it should not be possible for this action to succeed at both units.

8.2.1.2 Radiation Hazards

Operator actions included in the Level 1 PRA are generally performed in the main control room (MCR), which is protected from excessive radioactive contamination. In the current model, if the MCR ventilation/filtering system fails due to a seismic event in combination with a bypass or isolation failure, the MCR is assumed to be abandoned, and no post-core damage operator actions are credited. This case is quite rare. However, in a multi-unit accident, fission products could be present outside containment with much higher frequency.

8.2.1.3 Combustion

One plausible way for a single unit core damage accident to affect a second reactor at the site is if it causes a hydrogen explosion or fire that affects shared equipment/facilities (as happened at Fukushima, prior to core damage at the second and third units). At a PWR, it is possible for this to occur if combustible material escapes from containment into the auxiliary building, such as in a bypass accident or by degradation of penetration seals or isolation valves due to high pressures/temperatures (Bentaib, 2015; EPRI, 2021b). Combustible material could also come from the SFP following fuel degradation (EPRI, 2021b). This EPRI report recommends ventilation of the auxiliary building to limit buildup of combustible gases but notes the possibility of aerosol accumulation on filters that could cause a fire.

Appendix K.1 describes MELCOR sensitivity analyses examining the potential for combustion in the auxiliary building at the reference plant. One of these is an ISLOCA through the RHR piping, and the other is a station blackout with an isolation failure near the basemat and a 21 gpm RCP seal leak. Both cases cause buildup of hydrogen in parts of the auxiliary building, but the ISLOCA also discharges steam that inerts the atmosphere in that area. In the isolation failure case, combustion is predicted in the absence of ventilation, and it would be expected to cause failure of the auxiliary building. While ISLOCA scenarios are very rare, containment isolation failure is relatively likely following seismic events, at about 8 percent of seismic CDF (see

Table 3-16 in NRC [2023c]). While it is unknown what effect these combustion events might have on equipment at the opposite unit, physical separation within the shared auxiliary building is not adequate to rule out destruction of critical SSCs, resulting in inability to prevent core damage. The presence of large quantities of fission products in these scenarios might also have an impact on operator actions being performed in the auxiliary building.

In the more common scenario of containment overpressure failure at the first unit, it is plausible that a similar transfer of hydrogen could occur, though substantially delayed compared with an isolation failure. Liner tears due to overpressure are expected to occur at discontinuities such as hatches, penetrations, and the liner-basemat junction, and the reference plant's Individual Plant Examination identifies the basemat junction and equipment hatch as the most likely failure locations (see Appendix E in the Level 2 PRA report for internal events and floods [NRC, 2022b]). The expected failure locations would allow gases to escape either into the tendon gallery (as assumed for the late containment failure [LCF] source term) or the portion of the auxiliary building adjacent to containment (although the MELCOR model used for the L3PRA project combines this area with the rest of the building, dividing it only by level). Station blackouts have a lower probability of combustion in containment due to steam inerting, so the concentration of combustible gases can remain high at the time of overpressure failure. Steam condensation after entering the auxiliary building could result in a combustible atmosphere and a global deflagration that fails key equipment throughout the building.

In accidents with no containment failure, design basis leakage at a rate of 0.2 weight percent per day cannot transfer enough gas into the auxiliary building to cause combustion, though that might change with increased leakage rates due to high temperature and pressure.

8.2.2 Cross-Unit Basic Event Coupling Due to Common Cause or External Events

Most of the CCFs relevant to determining the RCs at the two units occur prior to core damage and so can either be carried over from the Level 1 PRA multi-unit calculations or are in the bridge tree. Possible post-core damage CCFs considered include operator actions and severe accident phenomena; these were determined to be generally independent across units. Operator actions are performed by separate crews, in separate locations, using separate equipment. The key phenomenological events are related to hydrogen combustion and thermally induced steam generator tube rupture, both of which are well understood mechanistically and have no plausible CCF mechanisms.

The current analysis of CCFs focuses on the two initiating events that will be quantified in the next section, weather-related LOOP (LOOPWR) and earthquake in seismic bin 6 (EQK-BIN-6). Top cutsets from these initiating events were examined to find candidate CCF events. LOOPWR was chosen because of its large contribution to internal events CDF and high multi-unit initiating event frequency. Seismic bin 6 was chosen for its potential to cause releases from multiple units as well as the spent fuel pool, making it an important contributor to multi-source risk.

8.2.2.1 LOOPWR

The coupling factors for LOOPWR (see Table 8-1) are all carried over from the MUCDF analysis. Events in the bridge tree and containment event tree were considered, and all the significant events were determined to be independent.

Table 8-1 Cross-Unit Coupling Factors for LOOPWR

Basic Event Name	Coupling Factor
1-IE-LOOPWR	0.625
1-ACP-CRB-CF-A205301	0.2
1-EPS-SEQ-CF-FOAB	0.2
1-EPS-DGN-CF-FRUN1	0.2
1-EPS-DGN-CF-FSUN1	0.2
1-EPS-MDP-FS-XFERPPS_-CC	0.2
1-SWS-MOV-CF-1668A69A	0.2
1-SWS-MOV-CF-116-ABCDEF	1
1-SWS-MDP-CF-FS-ABCDEF	1
1-AFW-TDP-FR-P4001___	0.2
1-AFW-TDP-FS-P4001___	0.2
1-OEP-XHE-XL-NR02HWR	1
1-OEP-XHE-XX-NR02HWR2	1
1-OEP-XHE-XX-NR02HWR1	1
1-OEP-XHE-XL-NR01HWR	1
1-OEP-XHE-XX-NR01HWR2	1
1-SWS-MDP-CF-FS-ABCDE (and all other combinations of 5/6 failures)	1
1-SWS-MDP-CF-FS-ABCD (and all other combinations of 4/6 failures)	1

The bridge tree consists of fault trees for the containment isolation system, containment spray system, and containment cooling system. The isolation system is irrelevant to this scenario, since the 1-REL-CIF frequency (see Table 8-3 for RC definitions) for the LOOPWR initiator is very small (i.e., 0.1% of total LOOPWR release frequency). The cooling and spray systems do contain CCF events, the most prominent being those shown below:

2.128E-4	1-CCU-MOT-FS-CCUALL__-CC	HIGH ORDER CCF COMB CAUSING CCU SYSTEM FAILURE TO START
1.048E-5	1-SWS-CTF-CF-FS-ALL	CCF OF 4 OR MORE (ALL COMBINATIONS) NSCW FANS TO START
1.120E-6	1-SWS-CTF-CF-FR-ALL	CCF OF 4 OR MORE (ALL COMBINATIONS) NSCW FANS TO RUN
4.878E-5	1-CSR-MDP-CF-START	CCF OF CS PUMPS TO START
1.338E-5	1-CSR-MDP-CF-RUN	CCF OF CS PUMPS TO RUN
1.187E-5	1-CSR-MOV-CF-HV9001AB	CCF OF CS PUMP DISCHARGE MOVs HV9001A & HV9001B TO OPEN

However, none of these events occur in the LOOPWR single-unit cut sets for 1-REL-LCF, 1-REL-ICF-BURN, or 1-REL-NOCF. Therefore, coupling factors for those events are not needed in this analysis.

SAPHIRE, system-generated CCF events (e.g. 1-SWS-MDP-CF-FS-ABCDE and other service water failures in the table above) can result in a large number of basic events that need coupling factors applied. The best way to handle these events is not entirely clear. In the calculations performed for the ISR task, each failure combination is coupled only with the same failure combination at the opposite unit (e.g., 1-SWS-MDP-CF-FS-ABDE and the corresponding Unit 2 event 3-SWS-MDP-CF-FS-ABDE[46] would be coupled to each other, but not to any of the other failure combinations).

Similarly, other CCF events can occur at one unit while a related event, but not the identical CCF, occurs at the other unit. Data is not generally available to calculate the correct dependence between these possible sets of failures. For example, in the multi-unit core damage cutsets for LOOPWR, there are cutsets that contain 1-EPS-SEQ-FO-1821U301 (Sequencer A fails to operate at Unit 1) and also 3-EPS-SEQ-CF-FOAB[47] (CCF of sequencers to operate at Unit 2). When considering both units together, this could be treated as a failure of 3 out of 4 sequencers, if an appropriate failure rate could be estimated. However, the potentially very large number of special cases would complicate the creation and slow the execution of the post-processing rules that apply coupling factors (see Section 12K.3). In the current calculations, all such asymmetric failure cutsets are left at their original probabilities.

8.2.2.2 Seismic Bin 6

The coupling factors for seismic bin 6 (see Table 8-2 below) are also mainly carried over from the MUCDF analysis (see Section 6.2.3.4.3 for the coupling factors used to develop MUCDF for seismic events). Most of these are 1.0 (including all seismic failures). A few used a dependent failure coefficient of 0.01 for MUCDF, and no coupling factor is assigned for these in the MURCF calculation because it would be minimally different from the independent failure probability. There are several new additions. 1-CIS-SYS-EQ6-ISO is a seismic failure in the bridge tree (seismic failure of containment isolation system) and so is treated as fully dependent between units. The others are basic events created from fault trees and are used in the single unit Level 2 PRA seismic model to simplify cutsets and reduce frequency inflation by replacing several events in a fault tree with a single probability (see Section 8.3.3 for more explanation of frequency inflation). These other basic events are 1-STRC-CD1-EQ6, 1-RCS-SLOCA-EQ6, and 1-BE-CISOL-EQ6-1.

[46] The convention adopted here is that Unit 1 events start with "1-" but Unit 2 events start with "3-," because the prefix "2-" was already in use for other parts of the model. See Section 12K.3 for details of the implementation.

[47] See previous footnote.

Table 8-2 Cross-Unit Coupling Factors for Seismic Bin 6

Basic Event Name	Coupling Factor
1-STRC-CD1-EQ6	1
1-RCS-SLOCA-EQ6	1
1-SWS-MDP-EQ6-P4_X	1
1-RCS-SYS-EQ6-AUXBLDG	1
1-EPS-DGN-2-EQ-6	1
1-SCC-SYS-EQ6-DCBS	1
1-SCC-SYS-EQ6-CBCHL	1
1-DC-125VBUS-1AD1-EQ6	0.2
1-DC-DPL-AD11-DD11-EQ6	1
1-EPS-MDP-EQ6-XFERP	1
1-DC-125VSWG-AD1-4-EQ6	1
1-DC-BAT-CHGR-EQ6	1
1-ACP-BAC-EQ6-4KV	1
1-AC-SEQDPL-A-B-EQ6	1
1-SSC-SYS-EQ6-NSCWTWR	1
1-RPS-ROD-EQ6-RCCAS	1
1-RCS-SYS-EQ6-CBLDG	1
1-EPS-DGN-EQ6-EXHAUST	1
1-ACP-INV-EQ6-POWER	1
1-AC-480VMCC-ABB-2-EQ6	1
1-EPS-DGN-EQ6-DGBLD	1
1-AC-480VBUS-BB16-EQ6	1
1-AC-480VMCC-1ABF-EQ6	0.1
1-AC-480VMCC-BBB-EQ6	1
1-FB-STR-EQ6-SYS	1
1-MSS-ADV-EQ6-PV30XX	1
1-ACP-TFP-EQ6-480V	1
1-ACW-HTX-EQ6-ACCW	1
1-CCU-SYS-EQ6-CUNIT	1
1-CCW-HTX-EQ6-SYS	1
1-AC-480VMCC-ABD-EQ6	1
1-SCC-SYS-EQ6-MCB	1
1-EPS-TNK-EQ6-DGDAY	1
1-RPS-ROD-EQ6-RTBRK	1
1-AFW-TDP-EQ6-CPNL	1
1-ACP-INV-VITAL-AC-EQ6	1
1-AC-480VMCC-ABF-2-EQ6	1
1-AC-120VPNL-AY2A-2-EQ6	1

Table 8-2 Cross-Unit Coupling Factors for Seismic Bin 6 (cont.)

Basic Event Name	Coupling Factor
1-RPS-SYS-EQ6-CONT	1
1-SCC-SYS-EQ6-SG	1
1-DC-125VMCC-EQ6	1
1-RCS-SYS-EQ6-MLOCA	1
1-AC-480VMCC-BBD-EQ6	1
1-EPS-DGN-EQ6-G400X	1
1-RPS-ROD-EQ6-RXV	1
1-BE-CISOL-EQ6-1	1
1-CIS-SYS-EQ6-ISO	1

1-STRC-CD1-EQ6 breaks out into a separate fault tree all the major structural failures that are considered independent (that is, they do not affect the bridge tree or containment event tree), which makes them safe to lump together. It can appear in cutsets as either a success probability or a failure probability. 1-RCS-SLOCA-EQ6 similarly lumps two seismic failures so that the lumped fault tree probability appears only as a success event. If it fails, then the particular basic event that caused its failure appears instead. 1-BE-CISOL-EQ6-1 is the probability of isolation system failure (fault tree 1-FT-CISOL-F) with the specific flag set that applies when 1-STRC-CD1-EQ6 succeeds. Success of 1-BE-CISOL-EQ6-1 is substituted for 1-FT-CISOL-F when that fault tree succeeds, so it appears only as a success event, and is incompatible with seismic failure of the isolation system at the opposite unit. Similar fault trees exist for the containment spray and cooling systems, and for the isolation system in the case where 1-STRC-CD-EQ6 fails, but none of these were significant contributors to the bin 6 cutsets.

8.2.2.3 Success Probabilities for Dependent Events

If a system failure has a dependency between the two units, then success of that same system is also dependent. In the case of fully dependent failure events, the success events are also fully dependent. However, if the coupling factor for the failures is less than 1, it becomes more complicated. For example, consider two identical failure events at the two, identical units: A and B. Since they represent equivalent failures, they have the same basic event probability p, defined as $p = P(A) = P(B)$. The cross-unit coupling factor α for these two events is defined as:

$$\alpha = P(B|A) = P(A|B).$$

The corresponding success coupling factor would be $P(\neg B|\neg A)$. By the definition of conditional probability,

$$P(\neg B|\neg A) = P(\neg B \cap \neg A) \,/\, P(\neg A),$$

and

$$P(\neg B \cap \neg A) = 1 - P(A \cup B).$$

To find the probability of the union, from the principle of inclusion-exclusion,

$P(A \cup B) = P(A) + P(B) - P(A \cap B)$.

And to get the intersection, again from the definition of conditional probability,

$P(A \cap B) = P(B|A)P(A) = \alpha p$

Combining these, the following is produced:

$P(\neg B|\neg A) = (1 + (\alpha-2)p) / (1-p)$

This formula gives a coupling factor for the success events that depends not only on the original coupling factor, but also on the single-unit failure probability.

Figure 8-1 plots this success event coupling factor as a function of the single-unit failure probability, for several values of the original coupling factor. The dotted line is for the case where the two units are independent, so the success events are also independent and $P(\neg B)$ = 1-p. As the coupling factor α increases, the probability of success at one unit given success at the other unit gradually increases above its independent value; however, the figure makes it clear that the difference is minimal unless α is greater than 0.2.

8.3 Quantification of Multi-Unit Release Category Frequencies

This section presents the quantification approach and results for multi-unit release category frequencies. Section 8.3.1 provides the MURCF results for weather-related LOOP events, while Section 8.3.2 presents results for seismic bin 6 events. Section 8.3.3 addresses the important issue of frequency inflation in the calculations and presents methods to address it.

To define RCs for multi-unit accidents, the key difference is that although core damage occurs at multiple units, the amount and timing of radioactive releases may differ dramatically between units. The source term for the site as a whole is a combination of the source terms at each unit. Therefore, the multi-unit RCs are combinations of single unit RCs. In this case, a multi-unit RC consists of a pair of RCs, in which the order is irrelevant. Table 8-3 shows the single unit RCs and their descriptions (more detailed RC descriptions are provided in Table 2-19 of the Level 2 PRA report (NRC, 2022b). Table 8-3 also indicates whether each RC contributes to LRF and LERF.

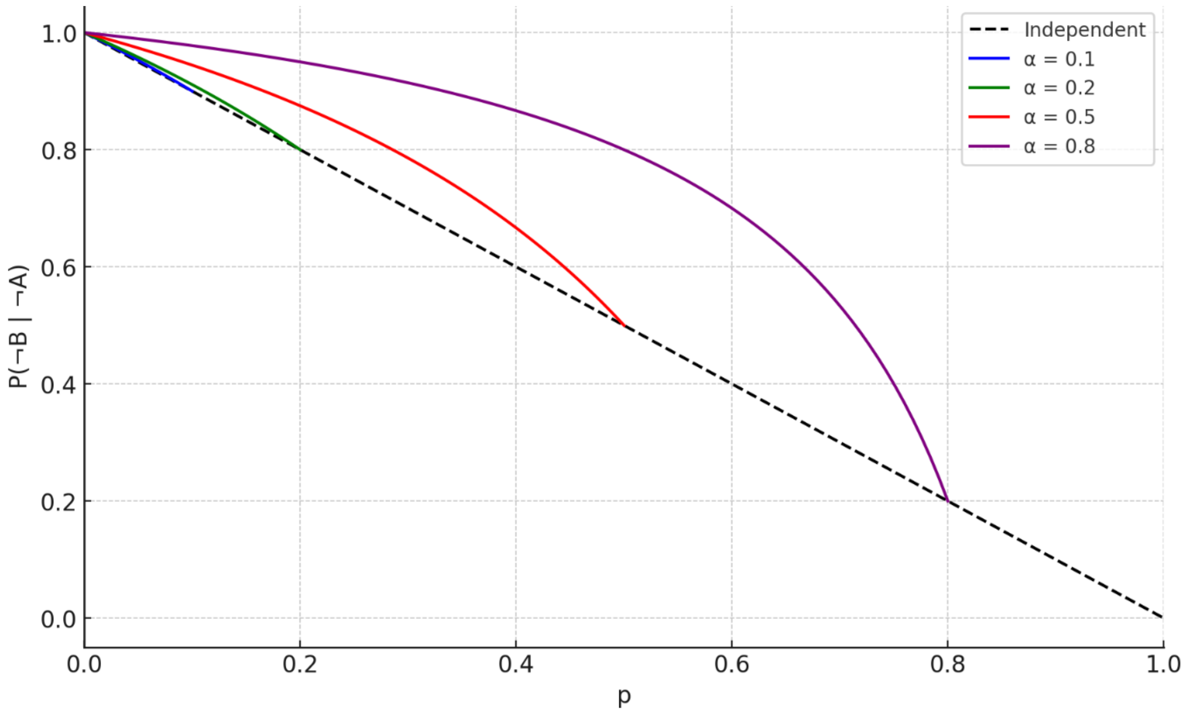

Figure 8-1 Success Event Coupling Factors

Since there are 16 single unit RCs, there are 136 possible unique pairings, such as BMT-CIF, ICF-ISGTR, and so forth. As can be seen later, most of these can be ignored due to low frequency, and a much smaller number of multi-unit RCs are chosen for quantification.

The quantification approach used for MUCDF could not be easily extended to MURCF calculations, because it relied on the observation/assumption that dual unit core damage scenarios would usually result from a CCF (in addition to the initiating event) that occurs at both units.

Table 8-3 Single Unit RCs for LOOPWR and Seismic Bin 6

Release Category	Description	LRF?	LERF?	LOOPWR Frequency (/rcy)	% of LOOPWR	EQK-6 Frequency (/rcy)	% of EQK-6
1-REL-BMT	Basemat Melt-through			1.65E-08	0%	2.38E-08	1%
1-REL-CIF	Containment Isolation Failure	Y		1.17E-08	0%	7.24E-07	20%
1-REL-CIF-SC	Scrubbed Containment Isolation Failure	Y		0.00E+00	0%	2.18E-08	1%
1-REL-ECF	Early Containment Failure	Y		1.34E-09	0%	5.96E-10	0%
1-REL-ICF-BURN	Intermediate Containment Failure due to Burn	Y		1.29E-06	12%	6.57E-07	18%
1-REL-ICF-BURN-SC	Scrubbed Intermediate Containment Failure due to Burn			1.23E-07	1%	7.71E-09	0%
1-REL-ISGTR	Thermally-Induced Steam Generator Tube Rupture	Y	Y	1.07E-07	1%	6.71E-08	2%
1-REL-LCF	Late Containment Failure by Overpressure	Y		5.00E-06	46%	1.45E-06	41%
1-REL-LCF-SC	Scrubbed Late Containment Failure by Overpressure	Y		1.94E-07	2%	2.17E-08	1%
1-REL-NOCF	No Containment Failure			4.02E-06	37%	4.98E-07	14%
1-REL-SGTR-C	PI-SGTR with Closed Secondary Side	Y		0.00E+00	0%	0.00E+00	0%
1-REL-SGTR-O	PI-SGTR with Faulted Secondary Side	Y		5.20E-11	0%	4.13E-08	1%
1-REL-SGTR-O-SC	Scrubbed PI-SGTR with Faulted Secondary Side	Y		5.56E-10	0%	4.85E-08	1%
1-REL-V	ISLOCA, auxiliary building intact	Y		0.00E+00	0%	0.00E+00	0%
1-REL-V-F	ISLOCA auxiliary building failed	Y	Y	0.00E+00	0%	0.00E+00	0%
1-REL-V-F-SC	ISLOCA, auxiliary building failed, break submerged	Y	Y	0.00E+00	0%	0.00E+00	0%
Total				**1.08E-05**	**100%**	**3.56E-06**	**100%**

Level 1 PRA multi-unit cutsets containing any random failures will generally have a much lower frequency of multi-unit core damage, since those random failures must occur independently at both units. This observation does not apply to Level 2 PRA failures—the entire MUCDF is allocated across a spectrum of RC combinations, so an identical or equivalent random failure does not need to occur in both units. Instead, a random failure might occur in one unit and not the other, causing them to end up in different RCs, but that cutset is still just as important to include. Additionally, many of the phenomenological basic events in the containment event tree have high enough failure probabilities that success terms are retained, so that even the very high frequency cutsets can contain random successes or failures. To get a good estimate of the frequency for each combination of two RCs requires generating combined cutsets. For illustration, take a simple example where after an earthquake a single basic event determines core damage, and a second one determines containment status. Then the Unit 1 cutsets would be:

1	1.2E-7	
	1.0E-6	IE-EQK
	0.3	1-STRC-FAIL
	0.4	1-CONT-FAIL
		End state: Containment failure
2	1.8E-7	
	1.0E-6	IE-EQK
	0.3	1-STRC-FAIL
	0.6	/1-CONT-FAIL
		End state: No containment failure

The Unit 2 cutsets would look similar. To find frequency of the multi-unit end state in which containment failure occurs at just one unit, a combined cutset is created using the corresponding basic events from both units:

1	2.2E-8	
	1.0E-6	IE-EQK
	0.3	1-STRC-FAIL
	0.4	1-CONT-FAIL
	0.3	2-STRC-FAIL
	0.6	/2-CONT-FAIL
		End state: Containment failure at Unit 1, intact containment at Unit 2

Since this refers to a specific combination (containment failure at Unit 1, intact at Unit 2) the frequency is then doubled to get the total frequency of a multi-unit core damage accident in which containment failure occurs at just one unit.

In actuality, there are multiple cutsets for each RC in the single unit model, so the multi-unit end state has a combined cutset for every pairing of Unit 1 and Unit 2 cutsets. The number of multi-unit cutsets is therefore roughly the number of Unit 1 cutsets times the number of Unit 2 cutsets, divided by 2, which can be well into the millions. Fortunately, many of these will fall below the truncation limit due to having a larger number of basic events per cutset. This may make this method useable even for plants with more than two units, depending on the flexibility to increase the truncation limit to reduce computational complexity. Cutsets that contain events with a cross-unit dependency are modified with SAPHIRE’s post-processing rules (see Appendix K.3 for details) to add a CCF event in place of the two units’ separate events, which generally causes them to become dominant contributors to both CDF and RCFs.

The number of multi-unit end states is proportional to the number of single unit end states to the power of the number of units. Due to the much greater number of possible two-unit end states relative to the 16 end states in the single unit model, it was necessary to limit this proof-of-concept analysis to the highest-frequency combinations of releases. Six of the 16 RCs are considered important for quantifying MURCFs, and there are 21 possible combinations of these 6 after removing duplicates (see Table 8-4).

This quantification method for the LOOPWR initiator was demonstrated using the RCs 1-REL-LCF (late overpressure failure), 1-REL-ICF-BURN (intermediate combustion failure) and 1-REL-NOCF (no containment failure), since these are by far the highest-frequency outcomes. (NOCF is included so that the total release frequency calculated will be comparable to the MUCDF.) Together these three categories make up 96 percent of single unit release frequency for LOOPWR, so the excluded combinations are not expected to contribute significantly to multi-unit release frequency. There are six possible combinations of these categories (highlighted in Table 8-4).

Table 8-4 Multi-Unit Release Category Combinations Considered for LOOPWR

	Unit 1 RC					
Unit 2 RC	**SGTR-O**	**ISGTR**	**ECF**	**LCF**	**ICF-BURN**	**NOCF**
SGTR-O	SGTROx2	ISGTR-SGTRO	ECF-SGTRO	LCF-SGTRO	ICF-SGTRO	NF-SGTRO
ISGTR		ISGTR-ISGTR	ECF-ISGTR	LCF-ISGTR	ICF-ISGTR	NF-ISGTR
ECF			ECF-ECF	LCF-ECF	ICF-ECF	NF-ECF
LCF				LCF-LCF	ICF-LCF	NF-LCF
ICF-BURN					ICF-ICF	NF-ICF
NOCF						NF-NF

For seismic bin 6, the release category 1-REL-CIF (containment isolation failure) is also important and was included in the calculation, bringing the total number of RC combinations to 10. Those four RCs make up 93 percent of single unit release frequency for seismic bin 6.

To generate the multi-unit cutsets, a new feature was implemented in SAPHIRE that allows conversion of a group of cutsets into a fault tree and then allows the fault tree to be duplicated while replacing the Unit 1 basic events with Unit 2 events. Then the Unit 1 and Unit 2 fault trees for their respective end states are linked with an AND gate in a third fault tree (Figure 8-2). Solving this third fault tree joins every pair of single unit cutsets to create the combined cutsets.

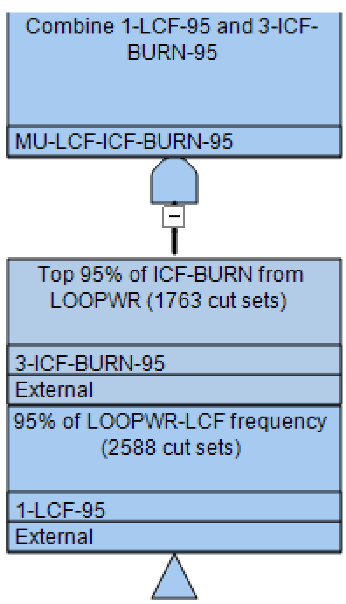

Figure 8-2 Fault Tree to Combine Cutsets from Different RCs at Two Units

For each of the two initiating events demonstrated here, this fault tree process was performed for each relevant RC combination. The results for LOOPWR and seismic bin 6 are provided in Section 8.3.1 and Section 8.3.2, respectively. An additional subsection, Section 8.3.3. discusses the issue of RCF inflation (i.e., when total release frequency in the Level 2 PRA exceeds total CDF from the Level 1 PRA).

8.3.1 Weather-Related LOOP MURCF Results

Frequencies of RC combinations for the LOOPWR initiator range from just 2 percent of MUCDF (ICF-BURN, ICF-BURN) to 48 percent (LCF, NOCF). Table 8-5 gives the frequency value and the truncation limit used for each multi-unit RC combination. The truncation limits were chosen by a convergence test, the results of which vary by RC (see Appendix K.3).

Although these combinations should theoretically capture slightly less than 100 percent of MUCDF (since low-frequency RCs were excluded), they instead sum to 127 percent of MUCDF. This is a result of frequency inflation, which can be greatly reduced by the same methods used for the seismic model (see Section 8.3.3 below for more discussion). However, this increase of approximately 30 percent is considered acceptable as-is, given the other substantial uncertainties in this calculation. There is substantial inflation between CDF and total release frequency in the single unit model as well, though inflation above 25 percent is normally only seen with seismic initiators.

The fault tree quantification method used for MURCF calculations was also used to calculate MUCDF for LOOPWR, to see how it compared to the LOOPWR MUCDF calculated using the cutset estimation method (CEM), described in Section 6. Using the fault tree method on cutsets representing about 96 percent of LOOPWR frequency, an MUCDF of 4.18×10^{-7}/rcy was obtained, quite close to the CEM value of 4.35×10^{-7}/rcy that was obtained using multiple cutset reviews, as discussed in Section 6.3.1.

Table 8-6 compares the MURCFs, calculated using the fault tree method to incorporate cross-unit dependencies, with the frequencies that would be expected if the RCs for the two units were completely independent. See Appendix K for details of this independent calculation.

Table 8-5 LOOPWR Multi-Unit Release Category Frequencies

MU Release Category Combination	Freq. of Combined Release (All Orderings) (/rcy)	Cutset Truncation Limit (/rcy)	% of LOOPWR MUCDF
ICF-BURN, ICF-BURN	8.371E-09	1E-19	2%
ICF-BURN, LCF	6.652E-08	4E-19	15%
ICF-BURN, NOCF	4.974E-08	1E-18	11%
LCF, LCF	1.312E-07	2E-18	30%
LCF, NOCF	2.100E-07	1E-18	48%
NOCF, NOCF	8.651E-08	1E-17	20%
Summed Release Frequency (for these 6 pairs)	**5.52E-7**		**127%**
LOOPWR MUCDF	**4.35E-7**		

Table 8-6 LOOPWR MURCF Comparison: Dependent versus Independent Calculations

MU Release Category Combination	Freq. of Combined Release (All Orderings) (/rcy)	% of Summed Release Frequency	Frequency of Combined Release if RCs are Independent (/rcy)	% of Summed Release Frequency if RCs are Independent
ICF-BURN, ICF-BURN	8.371E-09	1.5%	6.21E-09	1.6%
ICF-BURN, LCF	6.652E-08	12.0%	4.82E-08	12.1%
ICF-BURN, NOCF	4.974E-08	9.0%	3.88E-08	9.7%
LCF, LCF	1.312E-07	23.8%	9.38E-08	23.5%
LCF, NOCF	2.100E-07	38.0%	1.51E-07	37.9%
NOCF, NOCF	8.651E-08	15.7%	6.08E-08	15.2%
Summed Release Frequency (for these 6 pairs)	**5.52E-7**	**100.0%**	**3.99E-07**	**100.0%**

The dependent and independent frequency calculations appear very similar, with no clear trend toward correlation between the RCs at the two units. The frequencies differ mainly because of inflation. The similarities partly reflect that all the cross-unit dependencies included in the LOOPWR calculation are Level 1 basic events—events in the bridge tree and containment event tree were determined to be independent. Nonetheless, the Level 1 accident progression largely determines the state of the plant at the time of core damage and, therefore, the RC, so adding dependencies for Level 1 basic events would be expected to increase the frequency of combinations with the same RC for both units (i.e., LCF-LCF, NOCF-NOCF, and ICF-ICF) and decrease the frequency of combinations where the two units have different RCs. LCF-LCF and NOCF-NOCF do appear to be slightly increased, but ICF-ICF is slightly decreased. It is possible that the decrease is related to inconsistencies in cutset truncation, given the very low frequency of that RC combination. Note that, as stated at the beginning of Section 0, the truncation limits were chosen by a convergence test, the results of which varied by RC (see Section 12K.3 for additional details).

The similarity of the dependent and independent RCs is surprising in part because there is at least one identifiable mechanism that would seem to link the core damage mechanism to the RC. Most of the significant cutsets in the NOCF RC contain /1-L2-BE-MANUALTDAFWGEN, operator success at extending blind feeding of the turbine-driven AFW. This is only possible when the TDAFW has not failed in the Level 1 logic. Among the cross-unit dependencies for this LOOPWR calculation are 1-AFW-TDP-FR-P4001___ and 1-AFW-TDP-FS-P4001___, failure of the TDAFW to run and start, which are assigned coupling factors of 0.2. Since these failures are correlated, the frequencies of the NOCF category should be correlated as well. However, these two events do not appear in any of those blind feed cutsets (since a TDAFW failure would prevent blind feed). If the success probability were included, it would cause a small increase in the frequency of the NOCF-NOCF combination, but the failure probabilities are small enough to make retaining the success terms mostly irrelevant. Instead, cutsets in which TDAFW fails to start or run at both units would be spread across various combinations of the *other* RCs, LCF or ICF-BURN, where they are small enough that the effect is not noticeable.

Figure 8-3 compares the frequencies of initiating events, SUCDF and MUCDF, and multi-unit RC combinations for multi-unit LOOPWR, showing frequencies on a log scale. Although the single-unit and multi-unit initiating events have similar frequencies, multi-unit CDF is very rare compared to single-unit CDF due to the relatively weak coupling between the units, which also extends to the containment response. Figure 8-4 shows that the containment response for multi-unit CDF is divided fairly evenly into several combinations of late overpressure failure, intact containment, and intermediate combustion failure. As noted earlier, the containment responses at the two units are largely uncorrelated.

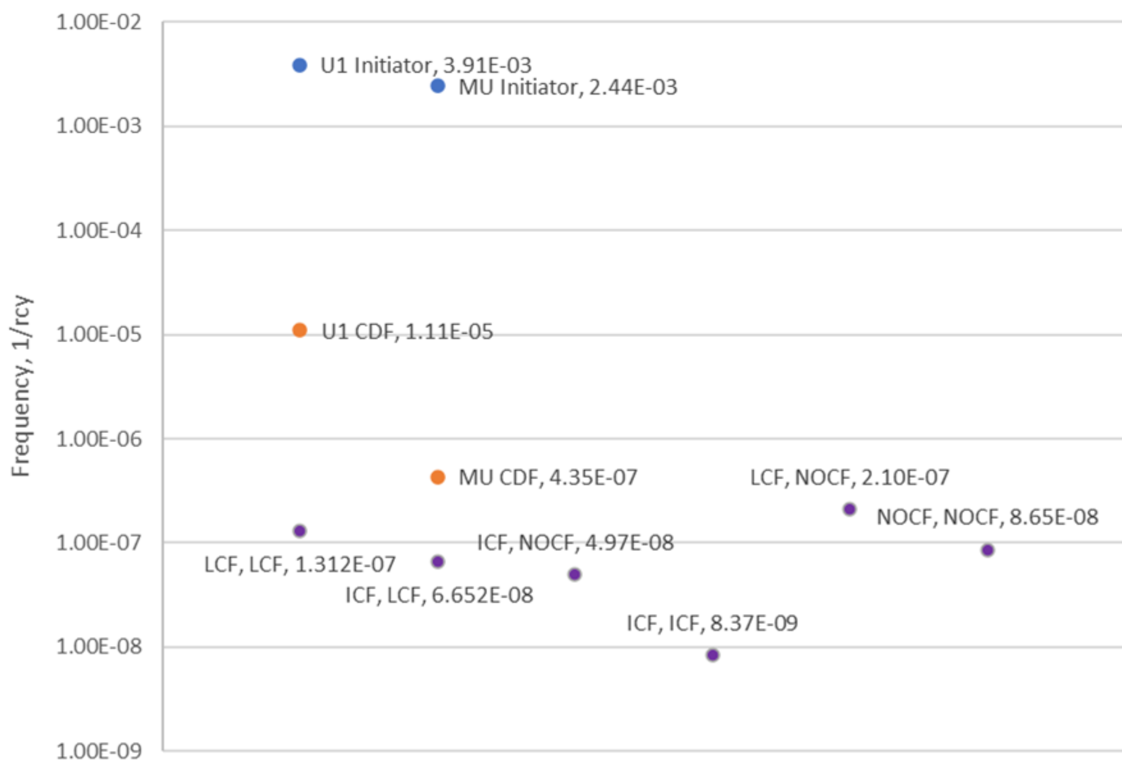

Figure 8-3 LOOPWR Frequency of Multi-Unit Core Damage and Key RC Combinations

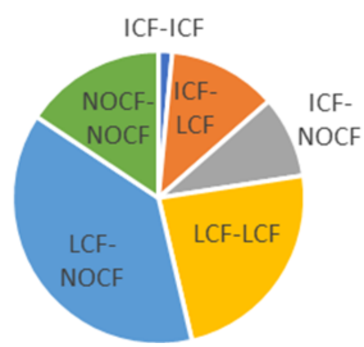

Figure 8-4 Frequency of LOOPWR RC Combinations

8.3.2 Seismic Bin 6 Results

The total multi-unit release frequency calculated in SAPHIRE for seismic bin 6 significantly exceeded the initiating event frequency, meaning that the degree of frequency inflation was excessive. The methods used to (partially) address this inflation are discussed in Section 8.3.3. The results presented here are the final values after adjustment.

Whereas the LOOPWR MUCDF is very similar for the CEM and the fault tree method, for seismic bin 6 the MUCDF calculated by the fault tree method is significantly lower (less conservative): a 14 percent reduction to 1.48E-6/rcy. This difference is largely due to the techniques used for controlling frequency inflation (discussed more below). A comparison of RCFs calculated by the fault tree method should only be directly comparable to the MUCDF calculated by the same method. However, even after adjustment for inflation, the total multi-unit release frequency calculated in SAPHIRE for seismic bin 6 significantly exceeds the original seismic bin 6 MUCDF calculated using the CEM approach.

For seismic bin 6, Table 8-7 gives the following information for each RC combination:

- frequency of combined release and percentage of total multi-unit release frequency for seismic bin 6 that would be expected if the RCs of the two units were completely independent (using the same method shown in Section 12K.2)
- frequency of combined release, accounting for coupling between the units
- cutset truncation limit
- percentage of total multi-unit release frequency for seismic bin 6, accounting for coupling between the units

These percentages with and without the independence assumption are not directly comparable—since the sum of the RCFs calculated by the fault tree method is inflated relative to MUCDF, it is significantly higher than the MUCDF calculated by either the fault tree method or the CEM approach (ideally, the sum of the RCFs should be slightly less than the total MUCDF, since only the largest combinations are included). Still, it is possible to observe general trends by comparing the relative percentages of the RC combinations for these two cases.

When the two units are treated as independent, every two-unit combination of the major seismic bin 6 release categories (CIF, ICF-BURN, LCF, and NOCF) has substantial frequency (second column of Table 8-7). This is simply the result of multiplying together their relatively high percentages of release frequency for seismic bin 6 in the single-unit PRA, which are shown in Table 8-8. For example, the 16.6 percent contribution of the CIF-LCF combination in Table 8-7 comes from multiplying the single-unit release frequency contributions of 20 percent for CIF and 41 percent for LCF, to get 8.3 percent, which is then doubled because it also includes the equivalent LCF-CIF combination where the roles of the two units are reversed. This approach is described in more detail in Section 12K.2 . Percentages of total SU release frequency are used, rather than percentages of SU CDF, because of the extreme frequency inflation (over 100 percent) in going from CDF to total release frequency. The total SU release frequency for seismic bin 6 is 3.56×10^{-6}/rcy, versus the SU CDF of 1.75×10^{-6}/ry.

Table 8-7 Seismic Bin 6 Multi-Unit Release Category Frequencies

MU Release Category Combination	Freq. of Combined Release if RCs are Independ. (/rcy)	% of Total MU Release Freq. if Independ.	Freq. of Combined Release (/rcy)	Cutset Truncation Limit (/rcy)	% of Total MU Release Freq.
CIF, CIF	7.106E-08	4.1%	2.772E-07	1E-15	11.2%
CIF, ICF-BURN	1.289E-07	7.5%	4.123E-10	5E-16	0.0%
CIF, LCF	2.847E-07	16.6%	1.202E-09	1E-15	0.0%
CIF, NOCF	9.786E-08	5.7%	3.550E-10	1E-15	0.0%
ICF-BURN, ICF-BURN	5.846E-08	3.4%	1.028E-07	1E-15	4.2%
ICF-BURN, LCF	2.582E-07	15.0%	5.394E-07	3E-15	21.8%
ICF-BURN, NOCF	8.876E-08	5.2%	2.067E-07	1E-15	8.4%
LCF, LCF	2.852E-07	16.6%	6.776E-07	1E-14	27.4%
LCF, NOCF	1.960E-07	11.4%	5.345E-07	2E-15	21.6%
NOCF, NOCF	3.369E-08	2.0%	1.315E-07	1E-15	5.3%
Sum	1.503E-06	87.4%	**2.472E-06**		100%
EQK bin 6 MUCDF (CEM)			1.72E-06		
EQK bin 6 MUCDF (fault tree method)			1.48E-06	4E-15	

Table 8-8 EQK-BIN-6 Single Unit Release Category Frequencies (Percentage of Single Unit Total Release Frequency)

Release Category	Percentage of Single Unit Total Release Frequency
BMT	1%
CIF	20%
CIF-SC	1%
ECF	0%
ICF-BURN	18%
ICF-BURN-SC	0%
ISGTR-EQ6	2%
LCF	41%
LCF-SC	1%
NOCF	14%
SGTR-C	0%
SGTR-O	1%
SGTR-O-SC	1%

Adding the effect of fully coupled seismic containment isolation failure events changes the distribution substantially. The cross-combinations where isolation failure happens at one unit and not the other, like the one just described, become extremely unlikely. The cutsets that do contribute to these RC combinations mainly involve isolation failure due to pre-existing maintenance error (1-L2-TEAR), which is unrelated to the seismic initiating event. Accounting for the coupling somewhat increases all other combinations because they absorb the frequency of those nearly eliminated pairings of one isolated unit and one unisolated.

Figure 8-5 compares the frequencies of initiating events, SUCDF and MUCDF, and multi-unit RC combinations for multi-unit seismic bin 6. In contrast to the LOOPWR results, here the single- and multi-unit CDFs are nearly identical, due to the high CCDP of this initiator. The RC combination frequencies span several orders of magnitude, with the smallest combinations (CIF, ICF and CIF, NOCF) not even visible on this plot. Figure 8-6 shows the release category combination frequencies as a pie chart, analogous to Figure 8-4 and demonstrating the increased importance of combustion failures and isolation failures for this seismic initiator.

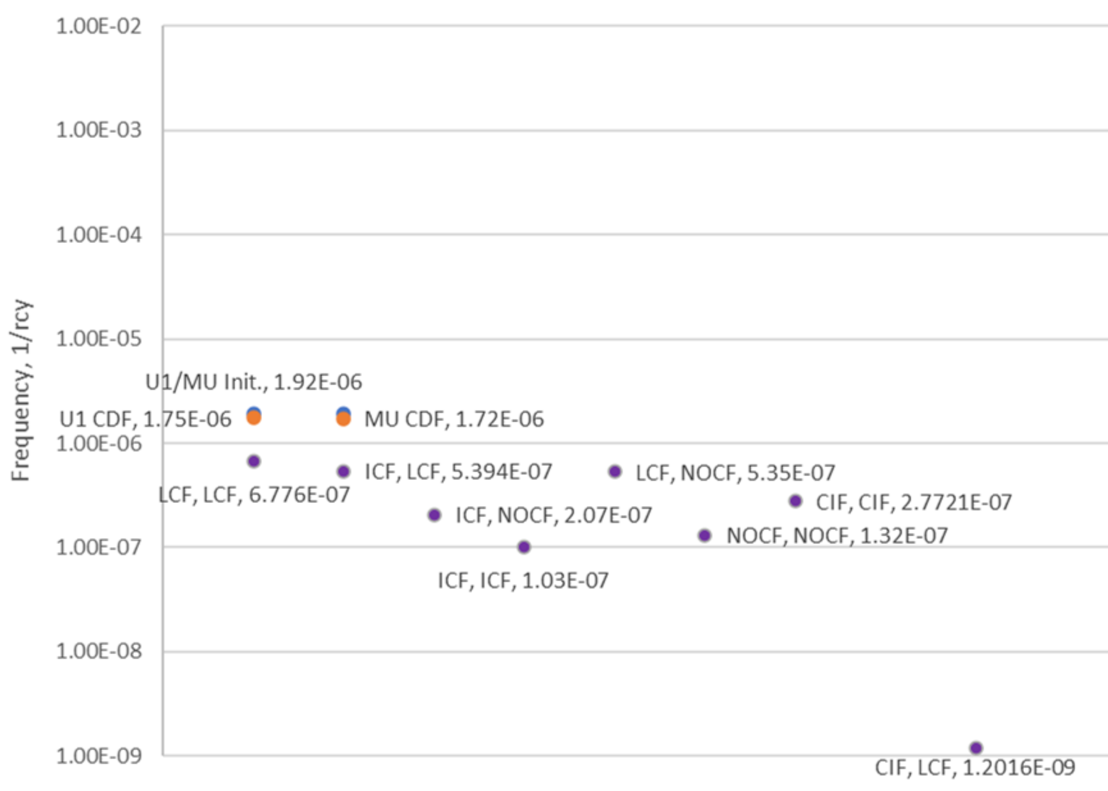

Figure 8-5 Seismic Bin 6 Frequency of Multi-Unit Core Damage and Key RC Combinations

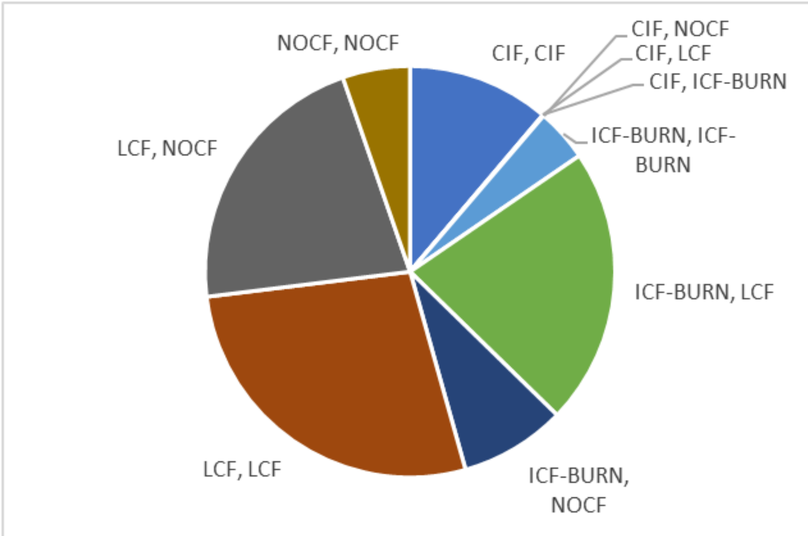

Figure 8-6 Frequency of Seismic Bin 6 RC Combinations

8.3.3 Frequency Inflation

In a perfect model, every core damage accident evolves to one RC, so the sum of RCFs should be exactly equal to the CDF. In SAPHIRE, though, it can be the case that the summed RCF results are much higher than the CDF for the same initiator, a problem referred to as frequency inflation.

One significant source of frequency inflation is the deletion of success terms in SAPHIRE's default quantification—it assumes that successes occur with a probability of approximately 1.0, which is often not the case following a severe seismic event when many failures have very high probabilities. That problem can be partially resolved by using the "I" or "W" process flag to add success events to the cutset results, thereby reducing the calculated probability of cutsets that include successes. The "I" process flag causes SAPHIRE to retain the success term for a particular event, or if applied to a fault tree, for all events in that fault tree. The "W" process flag can be applied to a fault tree to make SAPHIRE create a success event for the fault tree as a whole, rather than each individual basic event. However, a reduction in cutset probability does not always sufficiently reduce the corresponding end state probability, due to the minimal cutset upper bound (MCUB) approximation SAPHIRE uses to calculate the combined probability of the cutsets.

As an example, take four cutsets with two end states (assume initiating event frequency of 1.0):

1. A*/C -> ES1
2. B*/C -> ES1
3. A*C -> ES2
4. B*C -> ES2

Each of the basic events A, B, and C has a probability of 0.5.

The total conditional core damage probability (CCDP), combining all end states, should be:

$$P(A \text{ or } B) = A + B - A*B = 0.75$$

The MCUB calculation of the same probability is the same as the correct value, i.e.,

$$P_{MCUB}(A \text{ or } B) = 1\text{-}(1\text{-}A)(1\text{-}B) = 0.75,$$

In general, the two calculations are equivalent (i.e., 1-(1-A)(1-B) = A+B-A*B) whenever A and B are independent. However, MCUB can be higher when they are dependent (which happens, for example, if A and B are in cutsets that share at least one common basic event).

The end state probabilities in this example should be:

$$P(ES1) = /C*(1\text{-}(1\text{-}A)(1\text{-}B)) = 0.375$$
$$P(ES2) = C*(1\text{-}(1\text{-}A)(1\text{-}B)) = 0.375$$

In SAPHIRE's calculation, however, each cutset will be assigned probability 0.5 * 0.5 = 0.25, and when quantified with MCUB each end state will have the following probability:

$$P_{MCUB}(ES1) = P_{MCUB}(ES2) = 1 - (1 - 0.25)(1 - 0.25) = 0.4375.$$

In calculations with high probability events, especially those for seismic PRA, there is an alternative method that can be used for cutset quantification, a Binary Decision Diagram (BDD). BDD gives the exact solution rather than an upper bound. The problem is that the size of the binary decision diagram increases very rapidly with increased number of cutsets (e.g., exponentially increasing in the worst case). Its use is, therefore, limited to relatively small numbers of cutsets. SAPHIRE includes a BDD quantification option for fault tree cutsets, which in practice can be used for groups of up to about a thousand cutsets.

Another option that SAPHIRE offers is called the SAPHIRE Cutset Upper Bound Estimator (SCUBE) (see Smith [2016], for a description). In this case, SAPHIRE divides a group of cutsets in two, quantifying the highest frequency cutsets with BDD and the remainder with MCUB, and then it combines those two values again using MCUB. This can be helpful because the majority of the frequency inflation may be caused by a relatively small number of high frequency cutsets, while the much larger number of low frequency cutsets (which may be too many for BDD) are more suitable for MCUB quantification. However, SCUBE's ability to reduce inflation is limited by what portion of the total frequency can be captured in the BDD group. If 75 percent of the frequency goes into the MCUB group, that portion will be unaffected and the reduction in frequency compared to MCUB will be less than 25 percent. The multi-unit cutsets used in the current calculation did not allow more than about 20–25 percent of frequency to be captured in the BDD group.

Table 8-9 compares some of the quantification methods considered for the seismic bin 6 MURCF calculation, based on preliminary results (i.e., these results may not exactly match those provided in Section 8.3.2). The results reported for seismic bin 6 in Section 8.3.2 were calculated by the Factored MCUB method, which is described in more detail below.

Table 8-9 Options for Seismic Bin 6 Quantification of Multi-Unit Release Category Frequencies

Quantification Method	Selected MURCF Frequencies (ICF-ICF, ICF-LCF, LCF-LCF)	Total, as % of MUCDF for Bin 6 (1.72E-6), Should be < 100%	Drawbacks
1. SAPHIRE MCUB (the default cutset quantification method)	2.60E-7, 1.81E-6, 2.72E-6	279%	Excessive inflation
2. SAPHIRE SCUBE on MURCF cutsets (top 20% of frequency solved by BDD)	2.28E-7, 1.59E-6, 2.46E-6	249%	Excessive inflation, takes longer to do, crashes SAPHIRE if too many cutsets are included in the BDD
3. Independent calculation (use fraction of total release frequency from single-unit PRA for each unit)	5.85E-8, 2.58E-7, 2.85E-7	35%	Doesn't account for dependencies. Might amplify distortion due to differing inflation across RCs.
4. Normalize MCUB frequency by the average inflation factor from seismic bin 6 in the single-unit PRA	6.28E-8, 4.38E-7, 6.57E-7	67%	Distorts the risk profile, since inflation is much higher in some RCs than others. Does not eliminate inflation.
5. Factored MCUB: separate out common events and re-quantify (with Python script)	1.03E-7, 5.39E-7, 2.07E-7	49% (but recall from Table 8-7 that when the other combinations are added, the total is 143% of MUCDF)	Customized methodology, not as easily reproduced. Still does not fully eliminate the inflation.
Note: All options are starting from cutsets that represent only about 96% of RCF.			

8.3.3.1 Factored Minimal Cutset Upper Bound Quantification Method

Factored Minimal Cutset Upper Bound (FMCUB) is a variant of the MCUB approximation for cutset quantification, developed during the L3PRA project to mitigate frequency inflation by reducing the upper bound on the probability of a group of cutsets. The high-level description here is intended to explain the purpose and effects of the quantification algorithm, as well as how it differs from the usual approach in SAPHIRE.

FMCUB is currently performed by publishing the cutsets and running a python script. In the future, it may be possible to integrate this method as an option in SAPHIRE. The script's quantification method is a variant of MCUB in which basic events shared between multiple cutsets are used as multipliers for the group of cutsets, rather than for each cutset individually.

In general, greater complexity of the cutsets (more events per cutset and more overlap between them) leads to a worse MCUB approximation and more inflation. The intent of FMCUB is to break a cutset group into smaller groups that can be simplified (by factoring out a shared event) and simplifying them as much as possible prior to applying MCUB.

For example, to find the probability for ES2 in the example above, the first step was to factor out the shared basic event C and use MCUB to calculate the combined probability of just the two cutsets conditional on C:

$$P(A \text{ or } B) = 1 - (1 - A)(1 - B) = 0.75$$
$$P_{FMCUB}(ES2) = C * P(A \text{ or } B) = 0.5 \times 0.75 = 0.375$$

In this simple case, this approach generates the exact answer (and likewise for ES1).

Now consider a case where there are multiple shared events in the cutsets belonging to a particular end state:

1. A*B*E → ES1
2. A*C*E → ES1
3. D*E → ES1

In this case, there are at first two options: (1) the "E," which is in all three cutsets, can be factored out or (2) the "A,", which is in just two cutsets, can be factored out.

If "A" is factored out, the probability of the first two cutsets is calculated as:

$$P_{FMCUB}(1 \text{ or } 2) = A*[1 - (1 - B*E)(1 - C*E)]$$

And the total probability, using MCUB to combine that with the third cut set, would be:

$$\begin{aligned} PFMCUB(ES1) &= 1 - (1 - D*E)*(1 - PFMCUB(1 \text{ or } 2)) \\ &= 1 - (1 - D*E)(1 - A*(1 - (1 - B*E)(1 - C*E))) \end{aligned}$$

However, this is not the exact answer. If, instead, the "E" is first factored out, then the "A" can be factored out as well for just the group of cutsets that share that event:

$$P_{FMCUB\text{-}alt}(ES1) = E * (1 - (1 - D)*(1 - P_{AB \text{ or } AC}))$$

where

$$P_{AB \text{ or } AC} = A*(1 - (1 - B)*(1 - C)).$$

This factoring can be performed recursively any number of times, starting with the largest groups of cutsets sharing a basic event and then operating on subgroups that share another event, as long as some shared event remains within the group of cutsets after factoring. When the remaining cutsets are all independent, their probability is calculated by MCUB as it would be in SAPHIRE.

In some cases, it is not obvious which event to factor out first. For example,

1. A*B → ES3
2. A*C → ES3
3. A*D → ES3
4. D*E → ES3
5. E*F → ES3
6. E*G → ES3

For this case, there are three shared events: A (cutsets 1, 2, and 3), D (cutsets 3 and 4), and E (cutsets 4, 5, and 6). It is not possible to factor out all of them simultaneously; factoring out any one of those three creates a group with no shared events. So, either the “A” and the “E” can be factored out to produce:

$$P(ES3) = 1 – (1 – A^*P_{B\text{ or }C\text{ or }D})(1 – E^*P_{D\text{ or }F\text{ or }G})$$

or the “D” can be factored out to produce:

$$P(ES3) = 1 – (1 – D^*P_{A\text{ or }E})(1 – A^*P_{B\text{ or }C})(1 – E^*P_{F\text{ or }G})$$

The choice of which common event to account for is somewhat arbitrary, since none of them will give the exact answer. In the implementation used here, the first event to be factored out is the one for which the sum of the probabilities of the cutsets that contain it is greatest (i.e. the event with the highest Fussell-Vesely importance); so, the second option would only be taken if:

$P_3 + P_4 > P_1 + P_2 + P_3$ (cutsets containing D have greater probability than those containing A)

and,

$P_3 + P_4 > P_4 + P_5 + P_6$ (cutsets containing D have greater probability than those containing E)

The ranking of shared events uses the rare event approximation. Another approach would be to just start with the event that appears in the greatest number of cutsets; this variation generally results in a slightly higher end state frequency estimate, because factoring out events that are common among low-frequency cutsets may prevent factoring out other events that are in a smaller number of higher-frequency cutsets.

8.4 Discussion of Risk Significance

This section examines the risk significance of multi-unit accidents. Section 8.4.1 discusses implications for LERF, Section 8.4.2 addresses LRF considerations, and Section 8.4.3 explores broader implications for multi-unit sites.

As shown in Section 6, multi-unit core damage accidents are very low frequency compared to single-unit accidents, at least for this two-unit plant. Therefore, they cannot be an important contributor to risk unless the consequences are far worse than a single-unit accident. In this section, the impact of multi-unit releases in terms of surrogate risk metrics are considered. The results show that, on the contrary, health consequences are not much worse for a multi-unit accident than for a single-unit accident. Consequently, multi-unit risk is not a major contributor to risk of early fatalities or latent cancers for the reference site.

The question of multi-unit offsite consequences more broadly, including economic damages, will be considered in Section 9, but here two surrogate measures for health effects were addressed. If multi-unit releases were to substantially increase LERF or LRF compared to the single-unit PRA, that result would be suggestive of a step change in consequences and, therefore, a potential superlinear effect that could make multi-unit core damage accidents risk significant despite their relatively small frequency.

Assumptions about multi-unit source term

For this section, multi-unit source terms have been estimated by merely summing the source terms from both units. This approach is typically somewhat conservative (whereas the analysis described in Section 9 uses a more realistic approach). In a multi-unit accident for the reference site, the two releases are unlikely to occur at the same time, because the timing of the most common modes of containment failure, the ICF-BURN and LCF RCs, is essentially stochastic. Therefore, the radionuclides in a realistic combined source term would be released more gradually (compared to a simple sum of the two source terms), and it would have longer associated warning times. Certain combinations, such as ISLOCA at both units or containment isolation failure at both units, might result in releases at nearly the same time—in these unusual cases, summing the source terms is realistic rather than conservative.

8.4.1 LERF

No combination of two non-LERF RCs' source terms can meet the definition of LERF (a 3.5-hour delay from declaration of General Emergency to release of 1 percent of iodine, combined with eventual release of 4 percent of iodine (see Appendix D of NRC [2022b]), except possibly a combination of two SGTR-O releases that occur at the same time. This combination is implausible not only because SGTR-O usually results from a single unit SGTR initiating event, but also because there is a long delay between the initiating event and the start of core damage, and this would differ somewhat between units. Therefore, it is possible to conclude that site LERF is not increased by multi-unit accidents.

8.4.2 LRF

It is likewise clear that the site LRF cannot increase due to multi-unit accidents, because the non-LRF RCs have small enough source terms that no combination of two of them would reach the LRF threshold, which is a Cesium release fraction of 2.9×10^{-4} (see Appendix D of NRC [2022b]). The non-LRF RCs are scrubbed intermediate combustion failure (ICF-BURN-SC), basemat melt-through (BMT), and intact containment (NOCF), and the largest of these is approximately 5 times lower than the LRF threshold. However, it is possible for a multi-unit accident to reach the LRF threshold earlier in the accident, particularly in the case of the late containment failure RC. Two LCF releases occurring simultaneously at the two units would reach the LRF threshold approximately 10 hours earlier than a single unit LCF release (57 hours after start of accident versus 66 hours). Nonetheless, it is unlikely that this modest acceleration of the release would cause significant harm beyond that of a single unit LCF release.

8.4.3 Implications for Multi-Unit Sites

For the two-unit reference plant, the results indicate that multi-unit releases are not a major contributor to overall risk. However, this conclusion may not hold for plants with many units, such as some advanced reactors, or where additional safety features are shared among units.

The observations about LRF and LERF for the two-unit reference plant are not expected to be applicable, in general. Source terms for a plant with a larger number of reactors could more easily combine to create a release that has substantial offsite consequences, while any of those reactors individually might have minor effects that do not require an emergency response or environmental remediation. Tighter coupling of the accident progressions could also bring the releases more in line temporally, increasing the likelihood of large early release compared to a single unit. Cascading failures may also become more risk significant for a multi-unit site with tighter coupling of the reactors.

8.5 Potential Uncertainty

This section addresses uncertainty considerations. Section 8.5.1 discusses parameter uncertainty and its treatment. Section 8.5.2 presents several possible sensitivity analyses that could be performed to better understand the robustness of the results.

No uncertainty analyses or sensitivity cases were performed for the MURCF calculations. However, discussion is provided below for potential future analyses.

8.5.1 Parameter Uncertainty

Parameter uncertainty can be calculated for the fault tree cutsets using SAPHIRE's Monte Carlo sampling method. The coupling factor basic events should each be assigned beta distributions with mean (μ) equal to the nominal coupling factor. The second parameter (b in SAPHIRE's interface) can be set to $0.5^{x}(1- \mu)/\mu$ for a constrained non-informative prior. However, for this analysis to be most accurate, the most important cross-unit CCFs should each have their own coupling factor basic event, rather than using the same one for all failures with the same coupling factor.

Parameter uncertainty analysis was not performed for this example, in part because combining it with the FMCUB quantification method used for the seismic bin 6 cutsets would be computationally difficult.

8.5.2 Possible Sensitivity Analyses

The following sections identify potential sensitivity analyses that might be pursued in future work.

8.5.2.1 Automatic coupling of all CCF events

One concern about the methodology used here for multi-unit quantification is that the coupling factors selected were based on review of the high frequency cutsets in the single-unit PRA. It is possible that some cutsets not significant to a single unit can form combinations that, once they have appropriate coupling factors applied, are major contributors to multi-unit risk. To investigate this possibility, a sensitivity calculation could be performed for the LOOPWR initiator, in which all CCF events are automatically assigned a coupling factor of 0.2. The analyst would then be able to focus their effort on high frequency multi-unit cutsets, rather than single unit, in order to determine which coupling factors are necessary and which should be removed. The result would be to identify any candidate CCF events that would have a high impact if they were determined to have potential for cross-unit dependency.

8.5.2.2 Asymmetrical Combinations

Another potential sensitivity analysis would investigate coupling between a CCF event at one unit and a random failure at the other unit that is part of the equivalent CCF group. For instance, if a 2 of 2 failure occurred at unit 1, and a single component from that group failed at unit 2, then the combination could be treated as a 3 of 4 failure. Even without knowing the appropriate coupling factor for 3 of 4, this sensitivity analysis could estimate an upper bound on the effect of including it.

8.5.2.3 Depletion of Shared Resources

If both units challenge a single water source, some of the multi-unit cutsets generated by SAPHIRE for a particular combination of RCs will have success events for both units, which makes them invalid. It is possible to remove these cutsets using post-processing rules; however, there is no simple way to reallocate their frequency to the RC combination that would occur if the action failed at one of the units, even if that combination could be identified. However, it should be possible to estimate an upper bound on the magnitude of the effect by summing the frequencies of the cutsets involved.

9 MULTI-UNIT AND MULTI-UNIT-SPENT FUEL POOL LEVEL 3 PRA

This section describes the seventh and eighth steps in the overall ISR task. This section provides a description of the approach used to quantify multi-unit (MU) Level 3 risk and the results. In addition, this section provides results for the combination of MU risk with that from other relevant radiological sources (i.e., the spent fuel pools [SFPs]). Results of previous ISR tasks and prior L3PRA project PRAs were used to perform these steps.

9.1 Approach

A scoping study was performed to estimate consequences from a MU accident involving combinations of at-power source terms using the MACCS multi-source capability. The following set of release categories (RCs) was chosen to reflect a range of release characteristics in terms of both timing (e.g., declaration of general emergency (GE)) and release magnitude.

- ICF-BURN
- NOCF
- ECF
- LCF
- ISGTR
- SGTR-O

In addition, selected combinations involving the CIF RC (CIF–CIF; CIF–NOCF, CIF–LCF, and CIF–ICF-BURN) were evaluated for consistency with the Level 1 and Level 2 PRA MU analyses documented in Sections 6 and 8. It should be noted that no combinations of reactor at-power source terms with either reactor low-power and shutdown source terms or SFP source terms were evaluated. Evaluations of such combinations is left as a candidate for future work; however, the authors believe that the combinations selected can provide insights into a wide range of potential multi-source consequences. The complete set of combinations evaluated with MACCS are listed in Table 9-1.

The consequence results were generated for the superposition of source terms generated for the single unit analyses (as documented in NRC [2022c] for internal events and internal flood initiators and NRC [2023d] for seismic initiators) using the MACCS multi-source capabilities. Both units were assumed to be characterized by a middle-of-cycle (MOC) core burnup. The accidents were assumed to be initiated at both units at the same time and the accident progression at each unit was assumed to be independent (i.e., accident progression at one unit was not affected by the accident progression at the other unit). Implementation of emergency response plans were assumed to be triggered off the earliest declaration based on conditions at either Unit 1 or Unit 2 and were assumed to be unaffected by the fact that an accident was initiated at more than one unit. The scoping study considered both nominal evacuation scenarios and degraded evacuation scenarios.[48]

[48] The degraded evacuation model was developed for the seismic and high winds analyses and involves changes to the shielding factors and evacuation modeling relative to the internal events and floods analysis.

Table 9-1 Release Magnitude and Emergency Declaration Times for Single Unit RCs Along with Assumed Emergency Declaration Time and Calculated Release Magnitude for the MU Calculation

Release Category	Unit 1		Unit 2		MU	
	Cs Release (Ci)	GE (hr)	Cs Release (Ci)	GE (hr)	Cs Release (Ci)	GE (hr)
ICF-BURN–ICF-BURN	5.91E+05	3	5.91E+05	3	1.18E+06	3
ICF-BURN–NOCF	5.91E+05	3	1.36E+03	8	5.93E+05	3
LCF–ICF-BURN	1.77E+05	3	5.91E+05	3	7.69E+05	3
LCF–LCF	1.77E+05	3	1.77E+05	3	3.55E+05	3
LCF–NOCF	1.77E+05	3	1.36E+03	8	1.79E+05	3
ECF–ICF-BURN	2.93E+06	8	5.91E+05	3	3.52E+06	3
ECF–LCF	2.93E+06	8	1.77E+05	3	3.11E+06	3
ECF–ECF	2.93E+06	8	2.93E+06	8	5.86E+06	8
ECF–NOCF	2.93E+06	8	1.36E+03	8	2.93E+06	8
NOCF–NOCF	1.36E+03	8	1.36E+03	8	2.72E+03	8
ISGTR–ICF-BURN	1.70E+06	8	5.91E+05	3	2.29E+06	3
ISGTR–LCF	1.70E+06	8	1.77E+05	3	1.88E+06	3
ISGTR–ECF	1.70E+06	8	2.93E+06	8	4.63E+06	8
ISGTR–NOCF	1.70E+06	8	1.36E+03	8	1.70E+06	8
ISGTR–ISGTR	1.70E+06	8	1.70E+06	8	3.40E+06	8
CIF–ICF-BURN	6.31E+05	3	5.91E+05	3	1.22E+06	3
CIF–LCF	6.31E+05	3	1.77E+05	3	8.08E+05	3
CIF–NOCF	6.31E+05	3	1.36E+03	8	6.32E+05	3
CIF–CIF	6.31E+05	3	6.31E+05	3	1.26E+06	3
SGTR-O–ICF-BURN	4.59E+06	47	5.91E+05	3	5.19E+06	3
SGTR-O–LCF	4.59E+06	47	1.77E+05	3	4.77E+06	3
SGTR-O–ECF	4.59E+06	47	2.93E+06	8	7.53E+06	8
SGTR-O–NOCF	4.59E+06	47	1.36E+03	8	4.60E+06	8
SGTR-O–ISGTR	4.59E+06	47	1.70E+06	8	6.29E+06	8
SGTR-O–SGTR-O	4.59E+06	47	4.59E+06	47	9.19E+06	47

9.2 MU Consequences

The results of the MU calculations are shown in Table 9-2 (for nominal evacuation scenarios, such as those initiated by a loss of offsite power) and Table 9-3 (for degraded evacuation scenarios, such as those initiated by a seismic event). These are results conditional upon occurrence of the selected RC combination and do not consider the frequency of the event. The consequences selected for tabulation are a subset of the consequences discussed in NRC (2022c) and NRC (2023d) and include the population-weighted individual early fatality risk within 1 mile of the site boundary, the population-weighted individual latent fatality risk within 10 miles, the collective effective dose within 50 miles, and the offsite economic costs within 50 miles. These consequence measures are those typically used in consequence analyses supporting regulatory analyses and are considered adequate to demonstrate potential methodologies for MU risk assessment.

Table 9-2 and Table 9-3 show both the results calculated for a MU, multi-source release (denoted by the heading “MU”) and the sum of the results from the independent single unit, single source releases (denoted by the heading (“U1+U2”)). Plots showing the relationship between the multi-source and summed single source results are provided in Figure 9-1 (for early health effects), Figure 9-2 (for latent health effects), Figure 9-3 (for collective effective dose within 50 miles), and Figure 9-4 (for economic costs within 50 miles). The diagonal dashed line in these plots represents the line at which the multi-source release is exactly equal to the sum of the single unit, single source releases. Points above this line therefore represent cases for which the consequences for the MU, multi-source release are higher than the sum of the independent single unit, single source releases, and points below this line represent cases for which the consequences for the MU, multi-source release are lower than the sum of the independent single unit, single source releases.

For early health effects, the consequences for the MU, multi-source release could be either more than, less than, or equal to the sum of the independent single unit, single source releases. Individual datapoints are labeled in Figure 9-1 to facilitate identification of which combinations result in consequences that are greater than the sum of the individual source term consequences. The combinations for which the consequences for the MU, multi-source release are more than the sum of the independent single unit, single source releases are two combinations (SGTR-O–SGTR-O and ISGTR–ISGTR) where an identical release was modeled from each unit. In this case, the timing of the releases is identical, but the source term is effectively doubled. This likely results in super-additive early fatality consequences because of the non-linearity of early health effects at these very low individual early fatality risk levels. For all other combinations, the difference in timing (i.e., evacuation initiated when the first unit reaches GE conditions) is sufficient to lower the exposures from the second release to below threshold levels for early health effects.

For latent health effects, the consequences for the MU, multi-source release were uniformly less than or equal to the sum of the independent single unit, single source releases. This result is consistent with the sub-linearity of the population-weighted individual latent cancer fatality risk, which is effectively constrained by protective actions. Because most of the risk arises from the late phase, which is unaffected by the effectiveness of evacuation, the nominal and degraded evacuation series effectively overlie each other. A similar pattern is seen for the collective effective dose within 50 miles, although the degree of sublinearity is considerably less at higher individual risk levels.

For economic costs, the consequences for the MU, multi-source release were generally comparable to the sum of the independent single unit, single source releases. The consequences of the multi-source release ranged from as little as half of the sum of the independent single unit, single source releases to as much as 30 percent larger. These results are also dominated by late phase contributions and are therefore not sensitive to the timing of the GE declaration. Therefore, the nominal and degraded evacuation series effectively overlie each other.

Table 9-2 Conditional MU Consequences: Nominal Evacuation Scenarios

Release Category Combination	Individual Early Fatality Risk, 0–10 mi (cases/person)		Individual Latent Fatality Risk, 0–10 mi (cases/person)		Collective Total Effective Dose (person-rem/yr), 0–50 miles		Total Economic Cost, 0–50 mi (2015$)	
	MU	U1+U2	MU	U1+U2	MU	U1+U2	MU	U1+U2
ICF-BURN–ICF-BURN	0.00E+00	0.00E+00	7.99E-04	1.34E-03	6.27E+05	8.34E+05	1.04E+10	1.02E+10
ICF-BURN–NOCF	0.00E+00	0.00E+00	6.70E-04	6.82E-04	4.18E+05	4.20E+05	5.15E+09	5.15E+09
LCF–ICF-BURN	0.00E+00	0.00E+00	8.98E-04	1.28E-03	5.42E+05	6.02E+05	6.37E+09	5.94E+09
LCF–LCF	0.00E+00	0.00E+00	7.88E-04	1.22E-03	3.15E+05	3.70E+05	2.10E+09	1.64E+09
LCF–NOCF	0.00E+00	0.00E+00	6.14E-04	6.22E-04	1.86E+05	1.88E+05	8.26E+08	8.52E+08
ECF–ICF-BURN	0.00E+00	0.00E+00	9.69E-04	1.52E-03	9.16E+05	1.15E+06	2.04E+10	2.04E+10
ECF–LCF	0.00E+00	0.00E+00	1.02E-03	1.46E-03	8.31E+05	9.17E+05	1.64E+10	1.61E+10
ECF–ECF	1.79E-09	0.00E+00	9.83E-04	1.71E-03	9.71E+05	1.46E+06	2.50E+10	3.06E+10
ECF–NOCF	0.00E+00	0.00E+00	8.57E-04	8.69E-04	7.32E+05	7.35E+05	1.54E+10	1.53E+10
NOCF–NOCF	0.00E+00	0.00E+00	2.81E-05	2.86E-05	5.85E+03	6.50E+03	3.72E+07	6.84E+07
ISGTR–ICF-BURN	5.64E-08	5.64E-08	9.78E-04	1.50E-03	8.69E+05	9.90E+05	1.39E+10	1.35E+10
ISGTR–LCF	0.00E+00	5.64E-08	1.05E-03	1.44E-03	6.90E+05	7.58E+05	9.05E+09	9.21E+09
ISGTR–ECF	5.79E-08	5.64E-08	1.09E-03	1.69E-03	1.08E+06	1.31E+06	2.34E+10	2.37E+10
ISGTR–NOCF	5.64E-08	5.64E-08	8.40E-04	8.50E-04	5.74E+05	5.76E+05	8.40E+09	8.42E+09
ISGTR–ISGTR	4.07E-06	1.13E-07	1.01E-03	1.67E-03	8.30E+05	1.15E+06	1.40E+10	1.68E+10
CIF–ICF-BURN	0.00E+00	0.00E+00	9.56E-04	1.46E-03	7.32E+05	8.38E+05	8.99E+09	8.14E+09
CIF–LCF	0.00E+00	0.00E+00	9.30E-04	1.40E-03	5.42E+05	6.06E+05	4.05E+09	3.84E+09
CIF–NOCF	0.00E+00	0.00E+00	7.95E-04	8.08E-04	4.18E+05	4.24E+05	2.99E+09	3.05E+09
CIF–CIF	0.00E+00	0.00E+00	9.53E-04	1.59E-03	6.56E+05	8.42E+05	7.85E+09	6.04E+09
SGTR-O–ICF-BURN	0.00E+00	2.49E-07	9.17E-04	1.33E-03	7.86E+05	1.08E+06	2.03E+10	2.11E+10
SGTR-O–LCF	0.00E+00	2.49E-07	9.47E-04	1.27E-03	6.39E+05	8.51E+05	1.61E+10	1.68E+10
SGTR-O–ECF	0.00E+00	2.49E-07	1.04E-03	1.51E-03	1.03E+06	1.40E+06	2.90E+10	3.13E+10
SGTR-O–NOCF	0.00E+00	2.49E-07	6.39E-04	6.73E-04	5.40E+05	6.69E+05	1.55E+10	1.60E+10
SGTR-O–ISGTR	5.64E-08	3.05E-07	1.09E-03	1.50E-03	9.96E+05	1.24E+06	2.34E+10	2.44E+10
SGTR-O–SGTR-O	7.88E-06	4.98E-07	7.87E-04	1.32E-03	9.27E+05	1.33E+06	2.22E+10	3.20E+10

Table 9-3 Conditional MU Consequences: Degraded Evacuation Scenarios

Release Category Combination	Individual Early Fatality Risk, 0–10 mi (cases/person)		Individual Latent Fatality Risk, 0–10 mi (cases/person)		Collective Total Effective Dose (person-rem/yr), 0–50 miles		Total Economic Cost, 0–50 mi (2015$)	
	MU	U1+U2	MU	U1+U2	MU	U1+U2	MU	U1+U2
ICF-BURN–ICF-BURN	0.00E+00	0.00E+00	7.99E-04	1.34E-03	6.30E+05	8.36E+05	1.04E+10	1.02E+10
ICF-BURN–NOCF	0.00E+00	0.00E+00	6.70E-04	6.82E-04	4.19E+05	4.21E+05	5.15E+09	5.15E+09
LCF–ICF-BURN	0.00E+00	0.00E+00	8.98E-04	1.28E-03	5.43E+05	6.03E+05	6.37E+09	5.94E+09
LCF–LCF	0.00E+00	0.00E+00	7.88E-04	1.22E-03	3.15E+05	3.70E+05	2.10E+09	1.64E+09
LCF–NOCF	0.00E+00	0.00E+00	6.14E-04	6.22E-04	1.86E+05	1.88E+05	8.26E+08	8.52E+08
ECF–ICF-BURN	0.00E+00	0.00E+00	9.70E-04	1.52E-03	9.23E+05	1.16E+06	2.04E+10	2.04E+10
ECF–LCF	0.00E+00	0.00E+00	1.02E-03	1.46E-03	8.37E+05	9.22E+05	1.64E+10	1.61E+10
ECF–ECF	1.52E-07	0.00E+00	9.84E-04	1.71E-03	9.87E+05	1.47E+06	2.50E+10	3.06E+10
ECF–NOCF	0.00E+00	0.00E+00	8.58E-04	8.70E-04	7.38E+05	7.40E+05	1.54E+10	1.53E+10
NOCF–NOCF	0.00E+00	0.00E+00	2.81E-05	2.84E-05	5.85E+03	6.48E+03	3.72E+07	6.82E+07
ISGTR–ICF-BURN	5.59E-07	5.59E-07	9.91E-04	1.80E-03	8.81E+05	1.01E+06	1.39E+10	1.35E+10
ISGTR–LCF	6.15E-08	5.59E-07	1.06E-03	1.74E-03	7.02E+05	7.77E+05	9.05E+09	9.21E+09
ISGTR–ECF	6.15E-07	5.59E-07	1.39E-03	1.99E-03	1.11E+06	1.33E+06	2.34E+10	2.37E+10
ISGTR–NOCF	5.59E-07	5.59E-07	1.14E-03	1.14E-03	5.93E+05	5.95E+05	8.40E+09	8.42E+09
ISGTR–ISGTR	1.82E-05	1.12E-06	1.68E-03	2.26E-03	8.73E+05	1.18E+06	1.40E+10	1.68E+10
CIF–ICF-BURN	0.00E+00	0.00E+00	9.56E-04	1.46E-03	7.33E+05	8.40E+05	8.99E+09	8.14E+09
CIF–LCF	0.00E+00	0.00E+00	9.31E-04	1.40E-03	5.42E+05	6.07E+05	4.05E+09	3.84E+09
CIF–NOCF	0.00E+00	0.00E+00	7.95E-04	8.09E-04	4.18E+05	4.25E+05	2.99E+09	3.05E+09
CIF–CIF	0.00E+00	0.00E+00	9.54E-04	1.59E-03	6.56E+05	8.44E+05	7.85E+09	6.04E+09
SGTR-O–ICF-BURN	0.00E+00	2.08E-06	9.17E-04	1.87E-03	7.99E+05	1.11E+06	2.03E+10	2.11E+10
SGTR-O–LCF	0.00E+00	2.08E-06	9.47E-04	1.81E-03	6.48E+05	8.81E+05	1.61E+10	1.68E+10
SGTR-O–ECF	0.00E+00	2.08E-06	1.04E-03	2.06E-03	1.05E+06	1.43E+06	2.90E+10	3.13E+10
SGTR-O–NOCF	0.00E+00	2.08E-06	6.39E-04	1.21E-03	5.51E+05	6.99E+05	1.55E+10	1.60E+10
SGTR-O–ISGTR	5.59E-07	2.64E-06	1.39E-03	2.33E-03	1.02E+06	1.29E+06	2.34E+10	2.44E+10
SGTR-O–SGTR-O	3.02E-05	4.16E-06	1.96E-03	2.40E-03	9.89E+05	1.39E+06	2.22E+10	3.20E+10

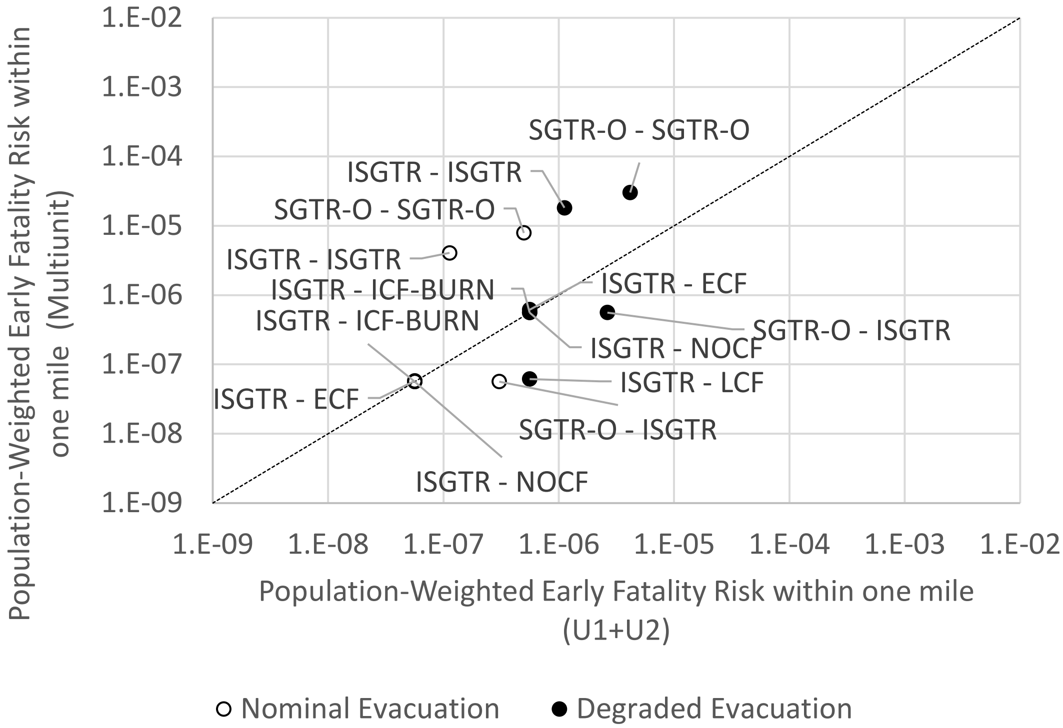

Figure 9-1 Early Health Effects: Multi-Source vs. Single Unit Results

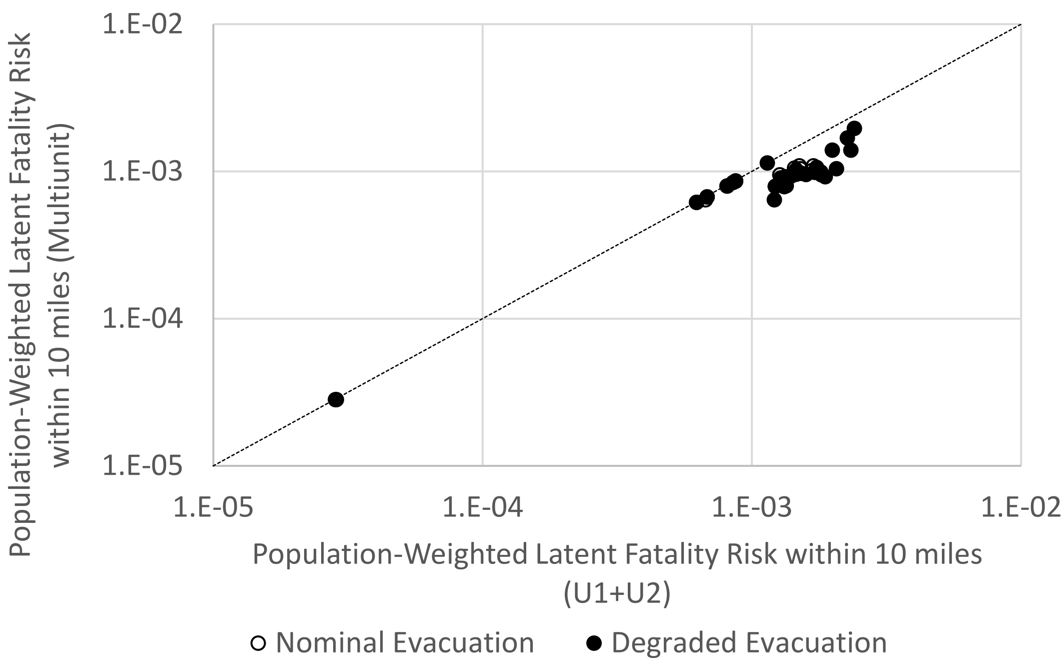

Figure 9-2 Latent Health Effects: Multi-Source vs. Single Unit Results

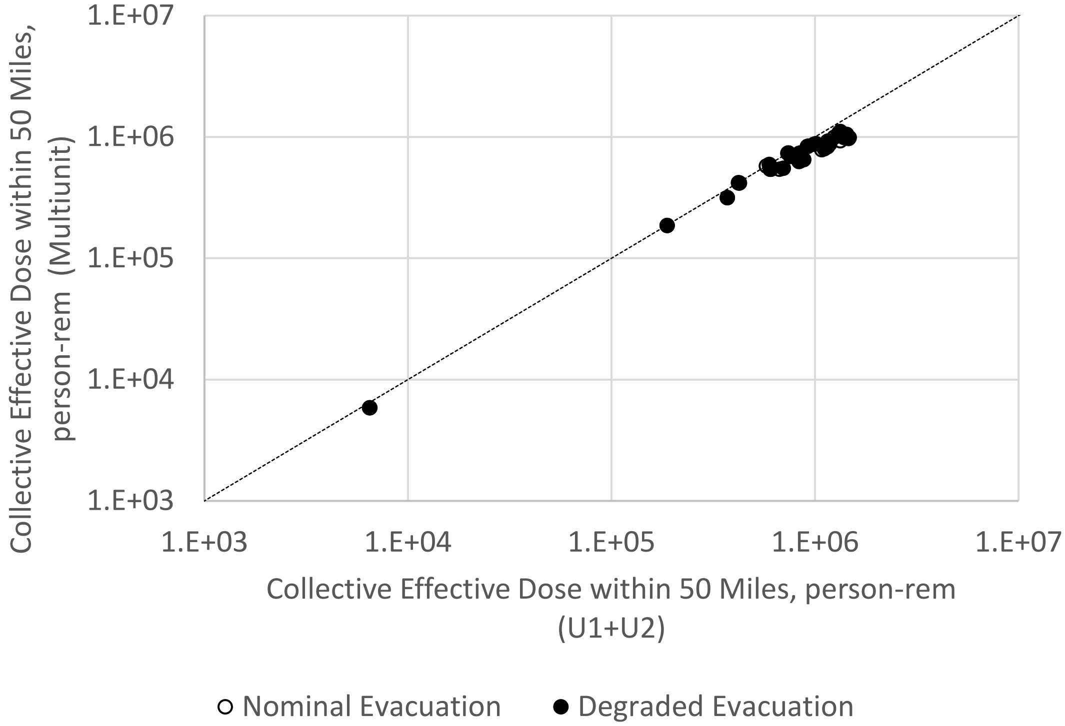

Figure 9-3 Collective Effective Dose within 50 Miles: Multi-Source vs. Single Unit Results

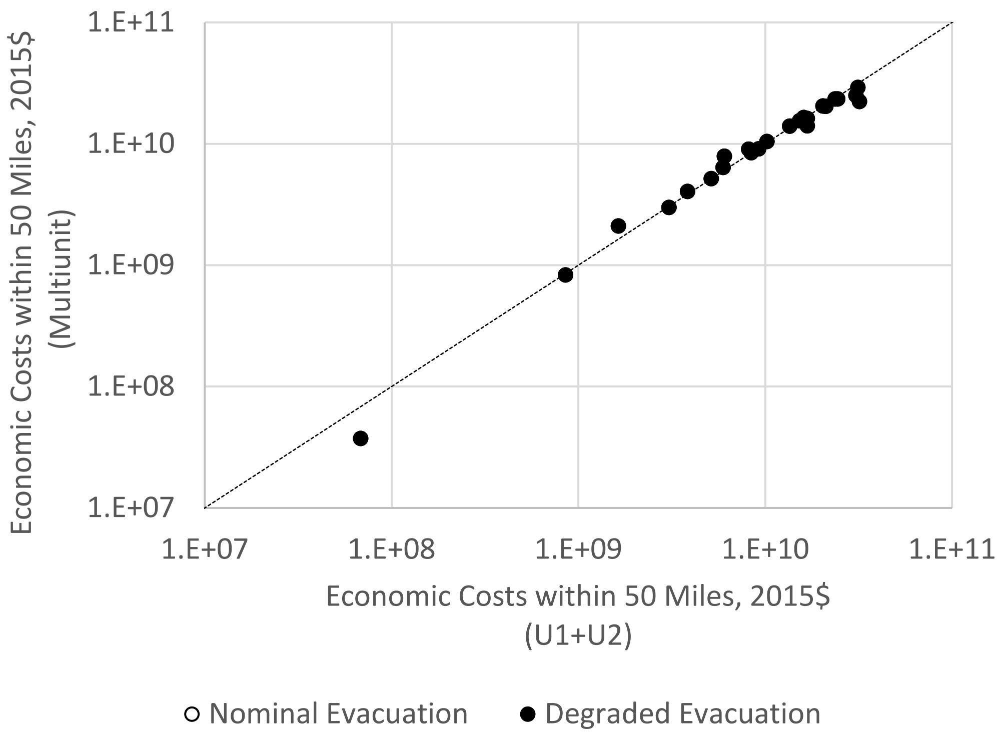

Figure 9-4 Economic Costs within 50 Miles: Multi-Source vs. Single Unit Results

These results demonstrate that the sum of consequences computed for independent source terms may be sufficient for estimating consequences from multi-source releases for some of the metrics of interest, although this may overestimate results for collective doses and latent health effects. This insight is consistent with the observation drawn from the single unit analyses that many consequences are either linear or sublinear with the magnitude of the release.

9.3 MU Level 3 Risk Integration

MU risk integration was performed by multiplying the frequency of RC combinations by the consequences computed as described in Section 9.2. MU RC frequencies (RCFs) that were reported in Section 8 (Table 8-5 and Table 8-7) are reproduced below in Table 9-4. This set of RC combinations is a subset of those identified in Table 9-1.

The results of frequency-weighting each RC combination are provided in Table 9-5. Because the frequencies identified above represent only a subset of initiating events—a weather-related loss of offsite power (LOOPWR) and a specific seismic event (EQK-BIN-6)—summation of frequency-weighted consequences for MU events were limited to only those with the same initiator. That is, the sum was taken over all LOOPWR events (with consequences taken from Table 9-2 for nominal evacuation scenarios) and overall EQK-BIN-6 events (with consequences taken from Table 9-3 for degraded evacuation scenarios). All RC combinations—except for the LOOPWR NOCF–NOCF and seismic bin 6 NOCF–NOCF, CIF–ICF-BURN, CIF–LCF, and CIF–NOCF RC combinations—contributed at least 5 percent of the risk to at least one consequence measure.

Table 9-4 MU RCFs for Selected Initiators

RC Combination	LOOPWR Frequency (/rcy)	EQK-BIN-6 Frequency (/rcy)
ICF-BURN, ICF-BURN	8.371E-09	1.028E-07
ICF-BURN, LCF	6.652E-08	5.394E-07
ICF-BURN, NOCF	4.974E-08	2.067E-07
LCF, LCF	1.312E-07	6.776E-07
LCF, NOCF	2.100E-07	5.345E-07
NOCF, NOCF	8.651E-08	1.315E-07
CIF, CIF		2.772E-07
CIF, ICF-BURN		4.123E-10
CIF, LCF		1.202E-09
CIF, NOCF		3.550E-10

Table 9-5 Frequency-Weighted MU Consequences

RC Combination	Release Freq. (/rcy)	Individual Latent Fatality Risk, 0–10 mi (/rcy)	Collective Total Effective Dose Risk (person-rem/rcy), 0–50 mi	Total Economic Cost Risk, 0–50 mi (2015$/rcy)
		LOOPWR		
ALL LOOPWR	***5.52E-07***	***3.34E-10[1] 100%***	***1.43E-01[1] 100%***	***1.22E+03[1] 100%***
ICF-BURN ICF-BURN	8.37E-09	6.69E-12 2%	5.25E-03 4%	8.70E+01 7%
LCF ICF-BURN	6.65E-08	5.97E-11 18%	3.60E-02 25%	4.24E+02 35%
ICF-BURN NOCF	4.97E-08	3.33E-11 10%	2.08E-02 15%	2.56E+02 21%
LCF LCF	1.31E-07	1.03E-10 31%	4.13E-02 29%	2.75E+02 23%
LCF NOCF	2.10E-07	1.29E-10 39%	3.91E-02 27%	1.73E+02 14%
NOCF NOCF	8.65E-08	2.43E-12 1%	5.06E-04 0%	3.22E+00 0%
		EQK-BIN-6		
ALL EQK-BIN-6	***2.47E-06***	***1.84E-09[1] 100%***	***9.40E-01[1] 100%***	***9.63E+03[1] 100%***
ICF-BURN ICF-BURN	1.03E-07	8.23E-11 4%	6.46E-02 7%	1.07E+03 11%
LCF ICF-BURN	5.39E-07	4.84E-10 26%	2.92E-01 31%	3.43E+03 36%
ICF-BURN NOCF	2.07E-07	1.39E-10 8%	8.65E-02 9%	1.07E+03 11%
LCF LCF	6.78E-07	5.34E-10 29%	2.14E-01 23%	1.42E+03 15%
LCF NOCF	5.35E-07	3.28E-10 18%	9.95E-02 11%	4.42E+02 5%
NOCF NOCF	1.32E-07	3.71E-12 0%	7.72E-04 0%	4.91E+00 0%
CIF CIF	2.77E-07	2.64E-10 14%	1.82E-01 19%	2.18E+03 23%
CIF ICF-BURN	4.12E-10	3.94E-13 0%	3.02E-04 0%	3.71E+00 0%
CIF LCF	1.20E-09	1.12E-12 0%	6.51E-04 0%	4.87E+00 0%
CIF NOCF	3.55E-10	2.82E-13 0%	1.48E-04 0%	1.06E+00 0%

Note 1: Results are a frequency-weighted sum of all RCs.

9.4 Multi-Source RC Combinations: MU–SFP Level 3 PRA Risk Integration

A scoping analysis was performed that included consideration of selected reactor at-power MU combinations coupled with SFP configurations where both reactors were at power (identified by an "AA*x*" operating cycle phase designator). The selected combinations, and their combined RCFs, are provided in Table 9-6. These values combine the seismic bin 6 frequencies for the multi-unit release categories "CIF * CIF" and "LCF * ICF-BURN" (from Table 8-7) with the conditional probabilities of SFP release for SFU5-AAN1, SFU5-AAN5, SFU6-AAN1, and SFU6-AAN5. These latter probabilities combine (multiply) the OCP fraction, the liner failure probability (for seismic bin 6), and the probability of the appropriate leak combination (i.e., probability of the appropriate leak size in each pool) as follows (note, all SPF-related values come from NRC [2025a]):

- SFU5-AAN1: 200 gpm leak in each pool (requires makeup strategy).
  Conditional failure = AAN1 OCP fraction * Liner failure probability (bin 6) * fraction for 200 gpm leak in pool 1 * fraction for 200 gpm leak in pool 2 = .09124 * 0.105 * 0.5 * 0.5 = 2.40E-3

- SFU5-AAN5: 200 gpm leak in each pool (requires makeup strategy).
  Conditional failure = AAN5 OCP fraction * Liner failure probability (bin 6) * fraction for 200 gpm leak in pool 1 * fraction for 200 gpm leak in pool 2 = .27007 * 0.105 * 0.5 * 0.5 = 7.09E-3

- SFU6-AAN1: 1500 gpm leak in pool 1, 2 or both (requires spray strategy).
  Conditional failure = 3 times the above number (3 options for leak combinations) = 7.19E-3

- SFU6-AAN5: 1500 gpm leak in pool 1, 2 or both (requires spray strategy).
  Conditional failure = 3 times the above number (3 options for leak combinations) = 2.13E-2

Table 9-6 Frequency-Weighted MU Consequences

RC combination	Combined RCF (/rcy)
CIF–CIF–SFU5-AAN1	6.65E-10
CIF–CIF–SFU5-AAN5	1.97E-09
LCF–ICF-BURN–SFU5-AAN1	1.29E-09
LCF–ICF-BURN–SFU5-AAN5	3.82E-09
CIF–CIF–SFU6-AAN1	1.99E-09
CIF–CIF–SFU6-AAN5	5.90E-09
LCF–ICF-BURN–SFU6-AAN1	3.88E-09
LCF–ICF-BURN–SFU6-AAN5	1.15E-08

*rcy – reactor-critical-year

For this analysis, explicit MACCS multi-source analyses involving SFP source terms were not available at the time of writing. Instead, drawing on the insights from Section 9.2, it was assumed that the multi-source RCs could be reasonably approximated by the summation of the reactor MU multi-source consequences and the SFP consequences listed in Tables 4.1 through 4.4 of NRC (2025b). Because all RC combinations were assumed to be initiated by a severe seismic event, the MU multi-source consequences from Table 9-3 (for reactor at-power multi-source consequences assuming a degraded evacuation) were used. These results are summarized below in Table 9-7.

Integrated reactor-SFP risk was estimated using the same approach used in Section 9.3; namely, by multiplying the frequency of RC combinations by the consequences computed in Table 9-7. The results of frequency-weighting each RC combination are provided in Table 9-8.

Because the frequencies identified in these tables represent only a subset of RC combinations, summation of frequency-weighted consequences for integrated reactor-SFP risk were limited to only the subset of RC combinations identified in Table 9-6. That is, the sum was taken only over the listed integrated reactor-SFP RCs. All RC combinations contributed at least 5 percent of the risk to at least one consequence measure.

Table 9-7 Reactor At-Power MU and SFP Consequences for Selected Multi-Source RC Combinations

RC combination	Individual Latent Fatality Risk, 0–10 mi (cases/person)			Collective Total Effective Dose, 0–50 miles (person-rem)			Total Economic Cost, 0–50 mi (2015$)		
	MU	SFP	MU +SFP	MU	SFP	MU +SFP	MU	SFP	MU +SFP
CIF, CIF, SFU5-AAN1	9.54E-04	1.20E-03	2.15E-03	6.56E+05	1.70E+06	2.36E+06	7.85E+09	4.30E+10	5.09E+10
CIF, CIF, SFU5-AAN5	9.54E-04	7.60E-04	1.71E-03	6.56E+05	4.20E+05	1.08E+06	7.85E+09	1.70E+09	9.55E+09
LCF, ICF-BURN, SFU5-AAN1	8.98E-04	1.20E-03	2.10E-03	5.43E+05	1.70E+06	2.24E+06	6.37E+09	4.30E+10	4.94E+10
LCF, ICF-BURN, SFU5-AAN5	8.98E-04	7.60E-04	1.66E-03	5.43E+05	4.20E+05	9.63E+05	6.37E+09	1.70E+09	8.07E+09
CIF, CIF, SFU6-AAN1	9.54E-04	1.20E-03	2.15E-03	6.56E+05	1.70E+06	2.36E+06	7.85E+09	4.30E+10	5.09E+10
CIF, CIF, SFU6-AAN5	9.54E-04	7.60E-04	1.71E-03	6.56E+05	4.20E+05	1.08E+06	7.85E+09	1.70E+09	9.55E+09
LCF, ICF-BURN, SFU6-AAN1	8.98E-04	1.20E-03	2.10E-03	5.43E+05	1.70E+06	2.24E+06	6.37E+09	4.30E+10	4.94E+10
LCF, ICF-BURN, SFU6-AAN5	8.98E-04	7.60E-04	1.66E-03	5.43E+05	4.20E+05	9.63E+05	6.37E+09	1.70E+09	8.07E+09

Table 9-8 Frequency-Weighted Multi-Source Consequences

RC Combination	Release Freq. (/rcy)	Individual Latent Fatality Risk, 0–10 mi (/rcy)	Collective Total Effective Dose Risk, 0–50 mi (person-rem/rcy)	Total Economic Cost Risk, 0–50 mi (2015$/rcy)
All reactor and SFP RCs	***3.10E-08***	***5.55E-11***[1] ***100%***	***4.11E-02***[1] ***100%***	***5.89E+02***[1] ***100%***
CIF, CIF, SFU5-AAN1	6.65E-10	1.43E-12 3%	1.57E-03 4%	3.38E+01 6%
CIF, CIF, SFU5-AAN5	1.97E-09	3.38E-12 6%	2.12E-03 5%	1.88E+01 3%
LCF, ICF-BURN, SFU5-AAN1	1.29E-09	2.71E-12 5%	2.89E-03 7%	6.37E+01 11%
LCF, ICF-BURN, SFU5-AAN5	3.82E-09	6.33E-12 11%	3.68E-03 9%	3.08E+01 5%
CIF, CIF, SFU6-AAN1	1.99E-09	4.29E-12 8%	4.69E-03 11%	1.01E+02 17%
CIF, CIF, SFU6-AAN5	5.90E-09	1.01E-11 18%	6.35E-03 15%	5.63E+01 10%
LCF, ICF-BURN, SFU6-AAN1	3.88E-09	8.14E-12 15%	8.70E-03 21%	1.92E+02 33%
LCF, ICF-BURN, SFU6-AAN5	1.15E-08	1.91E-11 34%	1.11E-02 27%	9.28E+01 16%

Note 1: Results are a frequency-weighted sum of all RCs.

9.5 Summary Observations

Context for the magnitude of MU risk compared to single unit risk is provided by Table 9-9, which shows the risk from all modeled RC combinations for MU LOOPWR, MU EQK-BIN-6, and multi-source risk (i.e., combination of MU and SFP) compared to the single unit risk estimates from the reactor at-power, reactor low-power and shutdown, and SFP analyses (summarized from Tables 5.3-1, 5.1-1, and 5.6-1 of NRC [2025b]). Caution should be used in this comparison, because the MU results do not include the full set of initiators analyzed in the single unit analyses.

In particular, Table 9-9 shows that:

- The risk from the full set of MU scenarios would have to be much higher than the MU LOOPWR and EQK-BIN-6 results in order to be comparable to the at-power single unit risk; but it is not for the reference plant.

- The single unit, low power and shutdown (LPSD) risk for internal events only is similar to that for the single unit, at-power, for internal events (and internal floods).

- The all-hazards, all modes (i.e., both at power and LPSD) risk results for the SFPs are roughly one or two orders of magnitude smaller than the corresponding (summed) single unit risk results.

- MU risk results for the two representative MUIEs are roughly one or two orders of magnitude smaller than the SU results.

- The results for multi-source risk for seismic bin 6 are substantially smaller than the MU risk for seismic bin 6.

Table 9-9 Summary of Single Source, MU, and Multi-Source Risk Measures

Scope Piece	Release Freq. (/yr)	Individual Latent Fatality Risk, 0–10 mi (/yr)[1]	Collective Total Effective Dose Risk (person-rem/yr), 0–50 miles[1]	Total Economic Cost Risk, 0–50 mi (2015$/yr)[1]
At-power (internal events and internal floods) for a single unit	6.9E-05	2.5E-08	9.9	80,000
At-power (all hazards[2]) for a single unit	1.6E-04	6.4E-08	27	230,000
LPSD (internal events) for a single unit	1.2E-05	4.5E-09	3.6	72,000
SFP (all hazards)	5.8E-07	6.3E-10	0.52	8,400
All MU LOOPWR[3]	5.52E-07	3.35E-10	0.14	1,220
All MU EQK-BIN-6[4]	2.47E-06	1.84E-09	0.94	9,630
Multi-source (i.e., simultaneous accidents for both reactors and SFP) for seismic bin 6	3.10E-08	5.55E-11	0.041	589

Note 1: Results are a frequency-weighted sum of all RCs. Reactor at-power results are based on reactor-critical-years. To convert the risk metric of the **reactor at-power** from per reactor-critical year to per calendar-year, multiply the result by the plant availability factor of 0.93. Note, the resulting reactor, at-power, calendar-year risk does not include the risk associated with reactor shutdown operations during the calendar year.

Note 2: Includes at-power internal events, internal floods, internal fires, seismic events, and high-wind events. Also, note that these values (and the other values reported in this table) do not include credit for the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Coping Strategies (FLEX), as well as other more recent plant changes, such as the installation of new reactor coolant pump shutdown seals. When these other changes are credited, the total release frequency from a single unit is reduced to 1.0E-04/rcy.

Note 3: LOOPWR – weather-related LOOP; the results shown are for simultaneous core damage at both units due to dependencies such as shared, connected, or identical structures, systems, and components [SSCs]).

Note 4: EQK-BIN-6 – seismic hazard bin 6; the results shown are for simultaneous core damage at both units due to their co-location on the same site during a seismic bin 6 earthquake.

10 KEY SOURCES OF UNCERTAINTY

The L3PRA project's ISR task is a proof-of-concept analysis. As such, there are many aspects of MU and multi-source risk that were not addressed in this report. These limitations, such as the scope limitations described in earlier sections of this report, can significantly influence the analysis results.

For example, the following scope limitations were identified as important in previous sections of this report:

- only at-power conditions addressed for the reactors
- only nominal conditions addressed for the SFPs (i.e., no operating cycle phases [OCPs] with a reactor shut down)
- FLEX strategies addressed in MUCDF results only
- only two MUIEs (i.e., weather-related LOOPs and seismic bin 6) addressed for MU Level 2 and Level 3 PRA
- a limited number of MU release categories (MURCs) addressed for MU Level 2 PRA
- only one multi-source scenario (i.e., seismic bin 6) addressed for combined MU and SFP risk
- only a limited number of MURC and SFP OCP combinations addressed in multi-source Level 3 PRA

In addition, because the ISR task relies on inputs from the L3PRA single source PRAs, all the same sources of uncertainty apply to the ISR task. Examples of sources of uncertainty that are specific to the ISR task include:

- MU coupling factors (e.g., MU CCF coupling factors)
- MU seismic correlations
- modeling approximations, especially for MU Level 2 PRA (e.g., see discussion in Section 8.5)
- lack of plant-specific information (e.g., plant drawings for the assessment of shared spaces)

11 SUMMARY OF FINDINGS

This objective of the L3PRA project's integrated site risk (ISR) task is to develop MU and multi-source risk results for two, nearly identical reactors, two hydraulically connected spent fuel pools (SFPs), and a dry cask storage (DCS) facility. The particular focus of the ISR task was on scenarios that could involve simultaneous consequences from two or more radiological sources.

These results summarized below include:

- Insights from overall results (Section 11.1)
- sitewide dependency assessment results for the two reactors, the SFPs with the two reactors, and the DCS with the two reactors (Section 11.2)
- at-power, multi-unit (MU) core damage frequencies (MUCDFs) for all hazards (Section 11.3)
- at-power, MU release category frequencies (MURCFs) for weather-related losses of offsite power (LOOPs) and seismic bin 6 events (Section 11.4)
- at-power, MU Level 3 PRA consequence and risk results for weather-related LOOPs and seismic bin 6 events (Section 11.5)
- the combination of at-power, MU Level 3 PRA consequence and risk results with "nominal" SFP Level 3 PRA consequence and risk results for an illustrative seismic bin 6 scenario (Section 11.6)

Given the various limitations of the current proof-of-concept ISR task, Section 11.7 identifies a few potential areas for future work.

11.1 Insights From Overall Results

This section provides insights for the overall ISR task. Some of these insights are discussed further in the sections below.

Regarding MU risk:

- Only certain IEs affect both reactors simultaneously (e.g., LOOPs, loss of service water, external hazards).
- The two reactors on the reference site were assessed to be very independent of each other. This independence is reflected in the MUCDF results, for example:
    - Total at-power MUCDF is only 10 percent of total at-power SUCDF.
    - The sum of at-power MUCDF for all four types of LOOP is less than 5 percent of the sum of at-power SUCDF for LOOPs.
    - The at-power, wind MUCDF is less than 1 percent of at-power, wind SUCDF.

- However, for seismic bins 5, 6, 7, and 8 (the highest seismic hazard bins), MUCDF is 90 to 100 percent of SUCDF.

- MU coupling factors (i.e., factors that represent cross-unit dependencies computationally) play a very important role in the calculated MU risks:

    - For weather-related and grid-related LOOPs, cutsets containing MU CCF coupling factors account for 86 percent of MUCDF.

    - For seismic bin 2, cutsets containing seismic hazards correlations (e.g., MU coupling factors) account for about 64 percent of MUCDF.

    - For seismic bin 6, cutsets containing MU seismic hazard correlations (e.g., MU coupling factors) account for 97 percent of MUCDF.

Regarding integrated MU and SFP risk:

- The two reactors and the two hydraulically connected SFPs were assessed to be independent, except for:

    - Seismic bins 5 and 6 are important contributors to both MUCDF and the SFP Level 1 and 2 PRA results.

    - Implementation of mitigation strategies for the reactors and SFPs share physical resources (e.g., portable pumps and water tanks) and associated human resources (e.g., operators).

- The L3PRA project team identified a unique multi-source scenario that involves nearly simultaneous (e.g., within the traditional 24-hour mission time) consequences at both reactors and SFPs. Specifically, the multi-source scenario is for a seismic bin 6 event that:

    - involves the sitewide dependencies mentioned above

    - can be mitigated but requires mitigation for SFPs within 24 hours, unlike most of the accident scenarios addressed by the L3PRA SFP PRA, where mitigation is not required for many hours after accident initiation

    - involves a timing dependency (i.e., given the amount of time it takes to complete the mitigative actions, they can be considered to occur in the similar timeframe):

        - the reactors require mitigation strategies to be completed by about 22 hours after reactor trip

        - the SFPs require mitigation strategies to be completed by about 10 hours after reactor trip

    - involves releases from both reactors and the SFPs

- For the illustrative seismic bin 6 scenario, the risk of the two reactors and SFPs combined is small compared to that for MU risk, for example:

- The single unit, low power and shutdown (LPSD) risk for internal events only is similar to that for the single unit, at-power, for internal events (and internal floods).

- The all-hazards, all modes (i.e., both at power and LPSD) risk results for the SFPs are roughly one or two orders of magnitude smaller than the corresponding (summed) single unit risk results.

- MU risk results for the two representative MUIEs are roughly one or two orders of magnitude smaller than the SU results.

- The results for multi-source risk for seismic bin 6 are substantially smaller than the MU risk for seismic bin 6.

There are several unique aspects of the L3PRA project's MU and multi-source risk analysis, including:

- To date, this is the only analysis that examined the combined risk of two reactors and two SFPs.

- A simplified approach was used to estimate MUCDF (which was justified, in part, by the lack of dependencies between the two reactors).

- Due to the lack of certain dependencies (e.g., shared SSCs) between the reactors and the SFPs, MU and SFP radiological releases could be added to serve as multi-source inputs for Level 3 PRA.

- As shown in the L3PRA single-unit Level 3 PRAs, it appears that source terms can be added to produce reasonable, yet conservative results as compared to those generated by the MACCS multi-source capability.

The L3PRA project's ISR task confirmed several state-of-the-art limitations, such as:

- If there are multiple reactors on site (i.e., more than two), the analysis of MU risk will get complicated very quickly. There will likely be numerous cross-combinations of cutsets and both the cutset estimation method (used in this study) and the more traditional event tree/fault tree approach may no longer be practical.

- Because there is limited data to support estimation of MU CCF coupling factors, conservative, generic coupling factors were used. As a result, the MUCDF estimates are expected to be conservative.

- Because there is little basis for MU seismic coupling factors, coupling factors were assigned using the SU seismic correlations and the expertise of the NRC's seismic PRA expert. Consequently, some of the MU seismic risk results may be conservative.

- As found in similar analyses, the L3PRA project's MU Level 2 PRA efforts were computationally challenging. Even for just two reactors, the number of MU RC categories that could be addressed was limited.

- Like other similar analyses of combined reactor-SFP risk (e.g., EPRI [2014]), the L3PRA project's ISR task addressed only at-power conditions. However, the single-source, low power and shutdown (LPSD) PRAs for the reactor and SFPs show that LPSD risk is significant.

11.2 Sitewide Dependency Assessment Results

The sitewide dependency assessment was performed for a range of dependency categories that was divided into three phases of assessment. Section 4 provides the details on how this assessment was performed as well as the results of the assessment. The high-level sitewide dependency assessment results are provided below:

- The two reactors are mostly independent except for some unavoidable dependencies, such as:
    - common initiating events (e.g., losses of offsite power and external hazards)
    - certain shared physical resources (e.g., electric power and water supplies)
    - certain shared systems, structures, and components (SSCs) (e.g., certain buildings, B.5.b pumps, and FLEX equipment)
    - identical components (i.e., MU common cause failures [CCFs])
    - hazard (e.g., seismic and high wind) correlations
    - human and organizational dependencies for certain operator actions credited in the single unit Level 2 PRA human reliability analysis
- The DCS facility is independent of the two reactors and is not included in sitewide risk estimates.
- For the SFPs in nominal conditions, the SFPs and the two reactors share resources (i.e., water supplies), equipment (i.e., B.5.b pumps), and operators for certain scenarios (e.g., seismic bin 6) that involve extensive damage mitigation guideline (EDMG) strategies, specifically:
    - external makeup strategy for the SFPs
    - containment spray cooling for the reactors in Level 2 PRA

These dependency assessment results were represented (e.g., via coupling factors or hazard correlations) in calculations for MU risk and the combined MU-SFP risk.

11.3 MUCDF Results

Section 6 addresses calculation of MUCDF, including how these results were produced. In particular, Section 6.3 provides the at-power, MUCDF results for all hazards, including treatment of both "base case" and "FLEX case."

One important result provided in Section 6.3 is that the total calculated MUCDF is approximately 10 percent of the total single unit CDF (SUCDF) developed in the traditional, single unit PRA

(SUPRA). However, the comparison of MUCDF to SUCDF is different for different MUIEs. For example:

- MUCDF and SUCDF are equivalent (or nearly equivalent) for seismic bins 6, 7 and 8.
- MUCDF is approximately 6 percent and 51 percent of SUCDF for seismic bins 1 and 3, respectively.
- MUCDF for each of the LOOPs addressed is generally between 1 and 5 percent of the parallel LOOP SUCDF.

Within the MUCDF results, the following insights were identified:

- The combination of contributions from all LOOP events to MUCDF is about 14 percent of the total MUCDF, with grid-related LOOPs contributing the most (about 8 percent of total MUCDF).
- Seismic events contribute over half of the total MUCDF (about 53 percent) with:
    - Bins 3 through 6 contributing almost 50 percent of the total MUCDF (nearly 94 percent of the total seismic MUCDF)
    - Bins 4, 5, and 6 contributing about 43 percent of the total MUCDF (over 80 percent of the total seismic MUCDF), in nearly equal shares
- The contribution from the loss of nuclear service cooling water (NCSW) initiating event is about 25 percent of the total MUCDF contribution.
- The contribution of wind-related events is 6 percent of the total MUCDF.
- The contribution from all four fire scenarios is about 2 percent of the total MUCDF.

The MU dependencies that underlie these results are different for different MUIEs. For example, MUCDF for LOOPs, loss of NSCW, wind events, and seismic bin 1 is dominated by MU CCFs. All other MUCDF results for seismic events are dominated by assigned seismic correlations.

11.4 <u>MURCF Results</u>

Section 8 discusses the approach for developing MURCFs and the associated results for two at-power, scenarios: weather-related LOOPs and seismic bin 6 events. As part of the MU Level 2 PRA, potential MU dependencies were re-assessed with respect to containment failures. Also, MU release categories (MURCs) were defined and 10 were selected to develop results (as discussed in Section 8.3).

For weather-related LOOPs, the following six MURCs were selected to develop MU Level 2 PRA results (with the respective percentage contributions to total MURCF for weather-related LOOPs):

- ICF-BURN, ICF-BURN = 1.5%
- ICF-BURN, LCF = 12%

- ICF-BURN, NOCF = 9 %

- LCF, LCF = 23.8%

- LCF, NOCF = 38%

- NOCF, NOCF = 15.7%

For seismic bin 6, a total of 10 MURCs were selected to develop MU Level 2 results, though only the following 6 MURCs were calculated to have non-zero contributions to total MURCF for seismic bin 6:

- CIF, CIF = 11.2%

- ICF-BURN, ICF-BURN = 21.8%

- ICF-BURN, LCF = 8.4%

- LCF, LCF = 27.4%

- LCF, NOCF = 21.6%

- NOCF, NOCF = 5.3%

In addition, MU large, early release frequency (LERF) and MU large release frequency (LRF) were discussed qualitatively, showing that multi-unit releases are not a major contributor to overall risk for the two-unit reference plant.

11.5 MU Consequences

Section 9 discusses the approach for developing MU consequences and the associated results for the MURCFs developed in Section 8 for two at-power scenarios: weather-related LOOPs and seismic bin 6 events. MU consequences were developed for four risk measures:

- individual early fatality risk
- individual latent fatality risk
- collective total effective dose risk
- total economic cost risk

In general, the MU consequence results demonstrated that the sum of consequences computed for independent source terms may be sufficient for estimating consequences from multi-source releases for some of the metrics of interest. This insight is consistent with the observation drawn from the single unit analyses that many consequences are either linear or sublinear with the magnitude of the release.

Section 9.5 provides some overall insights for these results as well as a caution regarding comparing MU and single unit risk (because the same set of initiating events were not analyzed). For example, the combined MU risk from all LOOPWR and EQK-BIN-6 events is substantially less than total single unit, reactor at-power risk. As noted in Section 9.5, these results indicate that:

- The risk from the full set of MU scenarios would have to be much higher than the MU LOOPWR and EQK-BIN-6 results in order to be comparable to the reactor at-power or low-power and shutdown scenarios; but they are not for the reference plant.

- The results for multi-source risk for seismic bin 6 are substantially smaller than the MU risk for seismic bin 6.

11.6 Multi-Source Scenario Consequences

Section 7 describes the illustrative multi-source scenario that was used to develop combined MU and SFP consequences. The specific details of this illustrative scenario allowed previously developed MU and SFP results to be combined.

The results for the multi-source (i.e., combined MU and SFP) consequences are given in Section 9.4. Eight combinations of MURCs and SFP operating cycle phases were addressed and results for three risk measures were developed:

- Individual latent fatality risk
- Collective total effective dose risk
- Total economic cost risk

Section 9.5 cautioned readers on comparing the results of the combined MU and SFP consequence results with those for single source results. However, the results given in Section 9 show that the integrated reactor and SFP risk is substantially less than either the reactor at-power, reactor low-power and shutdown, or spent fuel pool scenarios.

11.7 Potential Future Work

Since this report has documented the first time the NRC has performed an integrated site risk task, there are many potential future tasks that could be performed. Only a few are captured here:

- Addressing low power and shutdown conditions in future work is expected to be important for both MU risk and combined MU and SFP risk.

- Addressing plant sites for the existing fleet of NPPs that have more cross-unit dependencies would provide a broader understanding of MU risk.

- Addressing plant sites with more than two reactors would provide a broader understanding of MU risk.

- Addressing different plant designs (e.g., advanced reactors) that may have more cross-unit dependencies would provide a broader understanding of their potential risks.

- With further improvements in SAPHIRE, MUCDF calculations could be performed within SAPHIRE.

- Additional illustrative multi-source scenarios that involve both the reactors and the SFP would provide a broader understanding of multi-source risk.

- Additional information to support MU and combined MU and SFP risk calculation of FLEX scenarios would provide a more up-to-date understanding of multi-source risk.

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APPENDIX A
RISK EQUATIONS

This appendix is a collection of risk equations that are needed for explanations or calculations associated with the integrated site risk task.

A.1 Introduction

Because the integrated site risk task is about "risk," various equations, including Boolean algebraic equations, are needed to explain concepts or perform calculations.

A.2 Basic Equations to Understand Multi-Unit Risk

The first IAEA report on multi-unit probabilities safety assessment (IAEA, 2019) provides a good explanation of the key risk concepts for multi-unit risk.

First, some terms must be defined:

SUCDF = single-unit core damage frequency (traditional Level 1 PRA results)

SOCDF = single-unit ONLY core damage frequency

MUCDF = core damage frequency for both reactors

SCDF = site core damage frequency

Using these terms, the following is true:

SUCDF = SOCDF + MUCDF

Note that, for SOCDF, there are two types of accident sequences that must be captured:

- sequences involving IEs that impact only one reactor at a time
- sequences involving MUIEs but only one reactor goes to core damage

In turn, for MUCDF involving a two-unit site, there are two different types of accident sequences that must be captured:

- sequences involving MUIEs in which both reactors go to core damage
- sequences involving failures from one reactor propagating to the second reactor via some type of dependency

If the two reactors on site are totally identical (including dependencies between the reactors), then:

$$SOCDF_1 = SOCDF_2$$

$$SCDF = 2 \times SOCDF_1 + MUCDF$$

$$= 2 \times SUCDF_1 - MUCDF$$

A.3 Calculations of MUCDF Using CEM – Key Equations

Section 6 describes the cutset estimation method (CEM) approach used in the integrated site risk task for the L3PRA project to estimate MUCDF. Appendix I provides some additional details on these calculations.

A simple example is used to illustrate how the CEM approach is used to calculate MUCDF. The following terms are needed for this simple example:

U1-CS_1	Unit 1 cutset #1 (containing only an initiating event and a single CCF basic event)
U1-CS_1-CDF	core damage frequency contribution from Unit 1 cutset #1
CCF_1	Common cause failure basic event that appears in Unit 1 cutset #1
U1CDF	Unit 1 total core damage frequency for the initiating event appearing in Unit 1 cutset #1
U1IEF	frequency of initiating event appearing in Unit 1 cutset #1
MUIEF	multi-unit initiating event frequency associated with the initiating event appearing in Unit 1 cutset #1
CCF_1-CP	Conditional probability of Unit 2 experiencing the same CCF, given CCF_1
U1-CCDP	Unit 1 total conditional core damage probability for the initiating event appearing in Unit 1 cutset #1
MU-CS_1	Multi-unit core damage frequency contribution from Unit 1 cutset #1

In this simple example, Unit 1 cutset #1 contains only the SUIEF and CCF_1, that is,

$$\text{U1-}CS_1\text{-CDF} = \text{U1IEF} \times CCF_1$$

Since Units 1 and 2 are identical, their PRA models and associated cutsets are also identical (e.g., U1-CS_1-CDF = U2-CS_1-CDF and U1-CCDP = U2-CCDP). Based on this equivalence and the description above, the MUCDF contribution can be calculated by representing the dependent and random failures for Unit 2, adjusting for the MUIEF, and using the rare events approximation, that is,

$$\text{U1-}CS_1\text{-CDF} \times (\text{MUIEF/U1IEF}) \times [CCF_1\text{-CP} + \text{U1-CCDP} - (CCF_1\text{-CP} \times \text{U1-CCDP})]$$

where the MU scenario involving only the MUIEF and the same CCF occurring in both units is:

$$\text{MU-}CS_1 = \text{U1-}CS_1\text{-CDF} \times (\text{MUIEF/U1IEF}) \times CCF_1\text{-CP}$$

APPENDIX B
APPROACH FOR SITEWIDE DEPENDENCY ASSESSMENT

The material in this appendix was used to guide and assist Level 3 PRA (L3PRA) project task leaders in performing a sitewide assessment of dependencies in support of the integrated site risk (ISR) task. Parts of this appendix have been repeated in Section 4. However, no changes have been made to this appendix so that it can be used as stand-alone guidance for sitewide dependency assessment.

This assessment is an important beginning step in performing the ISR task. Dependencies between radiological sources (e.g., dependencies between the two reactor units) can complicate the integration of individual risk contributions. Consequently, the results of a sitewide dependency assessment can indicate how complicated the risk models that represent the site may need to be, as well as what interconnections between radiological sources need to be represented in such risk models.

The results of the sitewide dependency assessments are summarized in Section 4. Detailed results are provided in Appendix C through Appendix G.

How the results of sitewide dependency assessments are used is addressed in discussions of MU and multi-source estimations (see, for example, Sections 5 and 6). For example, once sitewide dependencies are identified and categorized, decisions were made to prioritize the identified dependencies by category or other measures. Such prioritization was necessary due to limited resources for performing the ISR task.

B.1 Background

For the L3PRA project, the ISR task estimated the risk contributions from modeled accident scenarios for the major radiological sources on the selected reference nuclear power plant (NPP) site, that is,

- two operating reactor units (Unit 1 and Unit 2)
- two spent fuel pools (SFPs), one for each operating reactor unit
- an independent spent fuel storage installation (ISFSI) or dry cask storage (DCS) facility

However, the individual contributions from the separate PRAs performed for the L3PRA project are not sufficient alone to perform ISR tasks. Sitewide dependencies must be identified and considered when estimating integrated risk for the entire NPP site. For example, accident scenarios that could involve core damage for both reactor units must be considered, as well as simultaneous failures of one or both reactor units and other radiological sources (e.g., one or both SFPs).

It should be noted that the overall freeze date for the L3PRA project is August 2012 (with a few exceptions that are documented in the respective L3PRA project reports). However, sensitivity analyses for FLEX strategies[49] have been performed for the two reactors, as documented in the single-unit PRA reports for the L3PRA project.

[49] 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

Other limitations in scope for the overall L3PRA project include:

- The L3PRA project only addresses reactor low power and shutdown risk for internal events, so the ISR task only addresses scenarios when both reactors are operating.
- The L3PRA project did not address seismically-induced fires quantitatively.
- Cross-unit internal floods for the control buildings were screened out.
- External floods were screened out.

These scope limitations can be considered as candidates for future research.

B.2 Prior Experience

Prior experience in calculating MU and multi-source risk is limited. Also, there is significantly less experience and development for multi-source than for multi-unit PRA (MUPRA). For example, recent reports by the IAEA (IAEA, 2019; IAEA, 2021a) and EPRI (EPRI, 2021a) address MUPRA but not multi-source risk. In addition, there is an on-going effort to develop a MUPRA standard. However, there is still much that needs further development, including the definition of basic terminology.

The sitewide dependency assessment guidance used in the L3PRA project is a blend of the IAEA guidance (IAEA, 2019 and 2021a) and the EPRI guidance (EPRI, 2021a) on sitewide dependencies. The guidance given in the IAEA and EPRI reports is focused on performing MUPRAs rather than multi-source risk. In particular, both IAEA and EPRI reports have a significance amount of guidance regarding the identification of sitewide dependencies between reactor units.

B.3 Categories of Sitewide Dependencies

The L3PRA project's ISR approach uses a categorization scheme to identify, characterize, and document the sitewide dependencies for the selected NPP site. This categorization scheme supports the systematic search for dependencies by recognizing the different ways dependencies can impact structures, systems, and components (SSCs) and operator actions. The specific categorization scheme that was used is a combination of similar schemes used in IAEA (2019, 2021a) and EPRI (2021a) for MUPRA.

Table B-1 (repeated from Section 4.2 in the main body of this report) provides high-level definitions of each dependency category, some illustrative examples, and expected ways that such dependencies could be represented in risk models.

The definitions and illustrative examples have been selected to guide analysts in assigning an identified dependency to a category because some types of dependencies might be interpreted to belong to multiple categories (i.e., some overlaps between categories may exist). Later sections provide brief descriptions of each category and guidance on their identification and representation.

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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.

The different categories of potential sitewide dependencies also are used to divide the assessment of sitewide dependencies into three phases, as discussed in the next section.

Table B-1 Potential Multi-Unit and Other Sitewide Dependencies

Category	Definition	Example(s)	How Modeled
Sitewide and Multi-Unit IEs	IEs that impact multiple reactor units and/or multiple radiological sources on site.	Loss of offsite power that are grid-, switchyard-, or weather-related.	Risk models are constructed to represent such IEs as causing reactor trip in both units concurrently.
Shared Physical Resources	Resources available to provide common support to reactor units, spent fuel pools, and dry cask storage facility.	Electric power via common grid and/or switchyard; ultimate heat sink, intake structure, water supplies for fire protection; diesel fuel, etc.	Common electrical grid and switchyard could be identified in this category but should be addressed under the category “sitewide and multi-unit IEs.” Other shared resources that could be identified (e.g., common water, diesel fuel) can be addressed in risk models.[1]
Shared or Connected SSCs	Shared or cross-tied systems and components that support multiple radiological sources under various conditions.	The service water system or a “swing” diesel generator may be shared by both reactor units and other radiological sources.	Like the “shared physical resources” of water and fuel, shared systems or components can be addressed in risk models (using flags or similar tools) after developing a scheme for prioritizing which radiological source is supplied first (or only).[1]
		For some NPPs, there may be an alternate alignment (e.g., cross-tie) of a system or component such that it can support the alternate unit (e.g., Unit 2 emergency diesel generator (EDG) can be cross-tied to feed Unit 1).[2]	Logic models and basic event naming for systems and components can be altered to account for cross-tied equipment.
	Shared or connected structures that support multiple radiological sources.	Two reactor units may share structures (e.g., turbine building) or may be connected (e.g., main control rooms for both units are connected).	Shared main control room (MCR) is a special case that should be addressed by human reliability analysis (HRA). Other shared or connected structures should be treated in a manner consistent with the hazard group (e.g., internal or external floods, internal fires, seismic event) that addresses the structures.
Identical Components	Components that have the same design, maintenance, operation, and operating environment for multiple units.	Failure of similar components installed in each unit due to common-cause.	Such dependencies can be addressed in risk models as potential inter-unit common-cause failures (CCFs).

Table B-1 Potential Multi-Unit and Other Sitewide Dependencies (cont.)

Category	Definition	Example(s)	How Modeled
Proximity Dependencies	Dependencies that arise across radiological sources from: (1) exposure of multiple SSCs to shared phenomenological or environmental conditions, (2) common features between units, or (3) operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source.	Failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat or cold, radiation levels), debris, explosions, etc., from a nearby radiological source. External hazard fails identical or similar structures due to common location of structures for both units.	These dependencies are not likely to have been identified in individual risk models for each radiological source. External hazards and radiological concerns (e.g., Level 2 PRA) are likely to be the principal concern for SSCs in both units that would share phenomenological or environmental conditions. Environmental conditions that impact operator actions can be modeled similarly to SSCs but should be addressed as "human or organizational dependencies.[3]" External hazards are likely to be of most concern for common features between units.
Human or Organizational Dependencies	Dependencies between operator actions across multiple radiological sources that can result from multiple causes, including sharing of staff and shared organizational factors.	Common training, procedures, human machine interface, or command and control structure cause recovery actions taken in response to an accident affecting one radiological source to be dependent upon those taken in response to an accident affecting another radiological source. Also, some staff (e.g., field operators, fire brigade, health physics, technicians) may be shared by all radiological sources on the site.	If dependencies related to common training, procedures, human machine interface, etc., need to be addressed, the HRA for each unit should be adjusted. Impacts from limited staffing, both in the Technical Support Center (TSC) and for field operators, can be represented by adjusting HEPs in the multi-unit model with Unit 1 getting full credit and Unit 2 receiving reduced or no credit.[4] It is not expected that the shared offsite technical support will be modeled.
Potential Accident Propagation Between Units	A reactor trip or subsequent failures on one reactor might cause an event for another radiological source on site.	Propagation of an accident from one radiological source to another is more likely if, for example, two units share systems or components, or are connected in some other way. Also, if conditions cause an automatic trip in one unit, then a manual trip in another.	Such dependencies may not have been identified in original PRA models.[5] Once they are identified, such dependencies can be addressed in logic models with, for example, Unit 2 failure being conditional on a certain Unit 1 failure(s).
Potential Hazards Correlations	SSCs and operator actions may be affected in the same or similar ways by external hazards (e.g., seismic or external floods).	Simultaneous (or nearly simultaneous) failures of SSCs or operator actions for both reactor units may occur due to impact of a seismic event.	Such dependencies are likely to be already addressed in the PRA for each radiological source and for the relevant hazard (e.g., the same seismic hazard curve and seismic correlations are used for both reactor units).

Table B-1 Potential Multi-Unit and Other Sitewide Dependencies (cont.)

Category	Definition	Example(s)	How Modeled
Notes: 1. Loss of these resources can be accommodated in the logic models for all impacted radiological sources by using the same basic event names in all the models. If these resources remain available, a priority scheme can be used for common supplies that designates reactor unit 1, for example, as getting all supplies it needs first, reactor Unit 2 as having secondary priority, and so on, until known supplies are depleted. Flag sets or similar PRA modeling techniques might be used to select which radiological source is credited with adequate physical resources (versus those that are not given such credit). Different time frames that are associated with different strategies (e.g., implementation of FLEX or Severe Accident Management Guidelines (SAMGs)) may imply different requirements and availability of physical resources. 2. The Level 2 HRA effort explored crediting the EDG cross-tie for the reference NPP but learned that: (a) it is not formally proceduralized, and (b) while operators and decision-makers are aware of this potential capability, it is not likely to be used since two reactor units could end up without AC power if the cross-tie is done improperly. Also, FLEX strategies and associated equipment are available now, making use of this option even more unlikely. 3. The timing of the conditions from one reactor that can affect another reactor is also important. Once such dependencies are identified as impacting SSCs or operator actions, one approach would be to assign conditional failure probabilities to basic events or HFEs (e.g., the basic events and HFEs in Unit 2's risk model can be altered due to failures, environmental conditions, debris, explosions, etc., that exist for nearby Unit 1). 4. The simple approach in existing guidance for MUPRA suggests that the HEP for Unit 2 actions be set to 1.0. 5. The EPRI guidance (EPRI, 2021a) suggests that such "cascading" dependencies only occur if there are cross-connected systems. Therefore, this category may not be important to the reference site as there are few such dependencies.			

B.4 Overall Approach for Sitewide Dependency Assessment

The assessment of sitewide dependencies required an in-depth understanding of the PRAs and underlying inputs related to the site and its radiological sources, support systems, interconnections, operations, and so on. This understanding was coupled with a systematic review of site-specific information (e.g., site layout drawings, system documentation, staffing plans, and procedures), to identify potential sitewide (i.e., inter-unit and inter-source) dependencies.

To meet the above-mentioned requirements, the ISR task's approach for identifying sitewide dependencies took advantage of the following resources:

- completed Level 1, 2, and 3 PRAs (models and results) for one reactor unit for internal events, internal floods, internal fires, seismic events, and high winds (the Level 3 PRAs for internal fires, seismic events, and high winds were currently under review, but preliminary versions were available for use)
- preliminary version of the combined Level 1 and 2 PRA (model and results) for the SFPs for multiple hazards (the Level 3 PRA for the SFPs was currently under development)
- preliminary version of the combined Level 1, 2, and 3 PRA (model and results) for DCS
- technical understanding of NPPs, SFPs, and DCS facilities and activities and their associated PRAs by L3PRA key technical leads

- site-specific emergency procedures (e.g., emergency operating procedures [EOPs], severe accident management guidelines [SAMGs], and FLEX procedures)

- recent guidance for performing multi-unit PRAs (MUPRAs) documented in IAEA (2019, 2021a) and EPRI (2021a)

This approach was also informed by on-going activities to develop a MUPRA standard. In addition, recent guidance on performing MUPRA provides important simplifications to the identification of sitewide dependencies that can make this task more efficient.

The L3PRA project's sitewide dependency assessment was performed by key technical leads using the guidance in this appendix. In some cases, brainstorming meetings were held with key technical leads, and led by the ISR task lead, to facilitate the sitewide dependency assessment. Follow-up activities were performed, as needed, to complete the assessments and their documentation. In some cases, candidate prioritizations were developed (e.g., risk importance measures or percentage contributions to core damage frequencies).

The assessment of sitewide dependencies was performed in a phased approach for the different categories of potential dependencies shown in Table B-1. Three phases were defined as follows:

- Phase 1 Assessment:

    - sitewide and multi-unit initiating events

- Phase 2 Assessment:

    - shared physical resources
    - shared or connected systems, structures, & components (SSCs)

- Phase 3 Assessment:

    - identical components (e.g., expansion of CCF groups)
    - proximity dependencies
    - human or organizational dependencies
    - accident propagation between units
    - potential hazards correlations

A phased approach offers three benefits:

- it focuses on the less difficult (and potentially more important) sitewide dependencies first

- it allows the use of earlier potential sitewide dependency results to inform the rigor of later sitewide dependency assessments

- it provides a general understanding of the "coupling" between reactors that determine the complexity of MU risk models that are built in later ISR tasks

Sitewide dependency assessment guidance for each phase is given in Sections B.5 through B.7 , and was generally implemented as follows:

Phase 1: Sitewide and multi-unit initiating events (MUIEs):

- First, review both reactor units and those initiating events (IEs) expected to be addressed in any MUPRA (based on IAEA [2019, 2021a] and EPRI [2021a]).

- Next, review the list of NPP initiating events used for the L3PRA project to identify any additional sitewide initiating events (using screening criteria provided in Section B.5).

- Finally, review these same initiating events and assess whether they also impact the SFPs and DCS.

Phase 2: Assessment of shared physical resources and shared or connected SSCs:

- First, consider information about the two reactors and identify any Phase 2 dependencies between them (using the categorization scheme given in Section B.3).

- Based on the results of the previous step, qualitatively assess the “coupling” between the two reactor units as “loosely coupled,” “tightly coupled,” or something in between.

- Then, review information about the SFPs and identify any Phase 2 dependencies between them and the reactors.

- If needed, review information about the DCS and identify any Phase 2 dependencies between it and either the reactors or the SFPs.

Phase 3: Assessment of remaining categories of potential sitewide dependencies, scaled by the results of Phase 2 dependency assessment (e.g., some dependency assessment can be bypassed if the Phase 2 dependency assessment for “coupling” between the two reactor units indicates “loose coupling”):

- First, consider information about the two reactors and identify any Phase 3 dependencies between them (using the categorization scheme given in Section B.3).

- Then, review information about the SFPs and identify any Phase 3 dependencies between them and the reactors.

- If needed, review information about the DCS and identify any Phase 3 dependencies between it and either the reactors or the SFPs.

In the ISR task’s process for identifying sitewide dependencies, the reactor Level 1 PRA for internal events was considered first, followed by external events, then Level 2 PRAs. A similar progression for the SFPs and DCS was followed. Since different analysts with different PRA type or hazard expertise were used in this process, some of these assessments were performed in parallel.

The most basic implementation of the steps above involves addressing only MUIEs and any important dependencies between radiological sources within the context of these MUIEs. Depending on the results of the sitewide dependency assessment, these contexts may be sufficient to represent simultaneous risk contributions from the radiological sources on site. For example, if the Phase 2 identification of sitewide dependencies results in few dependencies between the two reactor units, then the MU risk models can be developed more simplistically.

Due to resource limitations, this approach does not address:

- combinations of only one reactor unit with either the SFPs or DCS
- any plant operating states (POSs) beyond at-power operations

B.5 Guidance for Performing Phase 1 Sitewide Dependency Assessments

As noted above, Phase 1 dependency assessment addresses sitewide IEs (shown as the first category of dependencies in Table B-1), which is potentially the most important but also the easiest of the sitewide dependencies to model in an MU risk model. All MU plant sites will have MUIEs (and, likely, sitewide IEs).

IEs that impact multiple reactor units and/or multiple radiological sources on site are expected to be especially important to investigating MU and multi-source risk. In particular, the early identification of each IE can help to focus attention on only the relevant portions of the individual risk models that need to be incorporated into the integrated risk models. Sometimes this kind of IE is called a “common cause initiator (CCI).”

Sitewide IEs were identified through the following steps:

1. identify Level 1 PRA MUIEs for the two reactor units
2. identify MUIEs that also impact the SFPs
3. identify MUIEs that also impact DCS

Note that this approach first identifies initiating events that are important to the reactors, then assesses whether these same IEs are important to the SFPs and DCS. In other words, the focus of the ISR task is on scenarios involving both reactors and not on scenarios involving only one reactor and either the SFPs or DCS.

As described in Section B.4 for the overall sitewide dependency assessment, individual analysts who were most knowledgeable of the various L3PRA models (e.g., different PRA hazards and types) performed the reviews of the respective PRAs and associated materials. Consistent with other published guidance, lists of IEs used in Level 1 PRAs, and response to such IEs, are the focus of the Phase 1 sitewide dependency assessments.

The assessments for the reactors were performed in succession, each analyst building on the previous assessment. The order of inputted reactor results from the analysts was:

1. internal events Level 1 PRA for the two reactors
2. internal floods Level 1 PRA for the two reactors
3. fire, seismic, and wind Level 1 PRAs for the two reactors
4. Level 2 PRAs for the two reactors

In all cases, a Level 1 PRA single-unit IE should be retained for consideration as a MUIE if any of the “converse” screening criteria are true. The “converse criteria” are:

1. The IE immediately results in reactor trip in both units.

2. The IE immediately results in reactor trip of one unit and a degraded condition in the second unit.

3. The IE immediately results in degraded conditions in both reactor units.

A set of three worksheets (with some illustrative examples shown in red font text) were developed to guide and document the identification of MUIEs:

1. Table B-2 was developed to support assessment (and, if needed, plant-specific refinement) of pre-selected MUIEs using the converse screening criteria: (a) losses of offsite power (LOOPs) that are grid-related, switchyard-related, or weather-related; (b) seismic events; and (c) losses of shared support systems (e.g., losses of service water).

2. Table B-3 was developed to support the review of the list of IEs for the baseline, single unit PRA to identify any MUIEs in addition to those identified in Table B-2. This table also can be used to document why an IE was screened out from consideration as an MUIE.

3. Table B-4 was developed to support the identification (and refinement, if necessary) of MUIEs that affect the SFPs and (if needed) DCS, as well as both reactors.

Table B-2 Use of Converse Screening Criteria to Identify MUIEs for MU Risk: Level 1 PRA Results for a Single Reactor

Potential MUIE	Are Any of the Converse Criteria Met? (Yes/No) Which?	Refinement/Caveat Notes	PRA Results, Risk Significance, Other Notes	Screened Out?	
				Why?	Potential Negative Consequences of Screening Out IE?
LOOP (grid-related only)	*Yes (per converse screening criterion #1*				
LOOP (switchyard-related only)					
LOOP (weather-related only)					
Loss of NSCW		Under what conditions could both units be affected?			
Seismic events		For example, which “bins” (if any) match the criteria?			

Table B-3 Identify More Potential MUIEs from Reactor Level 1 PRA: IEs that Did Not Meet Converse Screening Criteria

Hazard Group	Number of IEs Reviewed/Source?	Number of IEs Screened Out	IEs that Survived Screening	Refinement/ Caveat Notes	PRA Results, Risk Significance, Other Notes
Internal Events[1]			*Other support system failures?*		
Internal Floods[1]					
Internal Fires			*MCR fire with abandonment*		
			Cable spreading room fire with MCR abandonment		
Seismic Events[1]			*Additional "bins" to those identified in Appendix A?*		
High Winds			*Tornado?*		*Screened out of single unit model because…?* *Identify as potential sensitivity case for ISR….?*
Other Hazards					

Notes:

1. Excluding those IEs already identified or confirmed as relevant MUIEs.

Table B-4 Search for Potential MUIEs that May Also Impact SFPs and DCS

L3PRA MUIEs	Relevant to SFPs/ DCS? (Y/N)	Refinement/ Caveat Notes	Level 1/Level 2 PRA Results, Risk Significance, Other Notes				Other Notes
			Risk Metric	% of total	Risk Metric	% of total	
LOOP (grid-related only)	N						LOOPs alone are not expected to be important to SFPs and DCS.
LOOP (switchyard-related only)							
LOOP (weather-related only)							
Seismic events	Y (see Notes)	*Bin 8*					Large seismic events are expected to be important to SFPs.
		Bin 7					
		Bin 6					
Loss of NSCW							

B.6 Guidance for Performing Phase 2 Sitewide Dependency Assessments

The assessment of potential sitewide dependencies in Phase 2 is important to determining the extent of coupling between reactor units on site. The EPRI report on MUPRA (EPRI, 2021a) states that tight coupling (i.e., multiple dependencies) between units requires more complex and quantitative risk modeling. On the other hand, if there is loose coupling between reactors (i.e., limited or no sharing of SSCs, limited or no connected structures), EPRI report (EPRI, 2021a) states that assessment of MU risk could consist of "…qualitative screening analysis and limited quantitative assessment of risk issues" that stem from sitewide dependencies. Phase 2 also is important in identifying any shared resources between the reactors and either the SFPs or the DCS facility.

The categories of potential sitewide dependencies (from Table B-1) that are assessed in Phase 2 are:

- shared physical resources
- shared or connected SSCs

Some of these types of dependencies can be difficult to assess (due to limitations in the availability of information or the state-of-the-art for PRA or hazard modeling). Therefore, it is recommended that analysts view the identification of such dependencies in a way similar to that for the identification of sources of uncertainty. In other words, the analyst should give their best effort to identifying such dependencies while recognizing that not all (and, maybe, only a few) of such dependencies can be represented in risk models. Also, the analyst should not be deterred from identifying a potential dependency if such a dependency is beyond the state-of-the-art for PRA or hazard modeling. The decision to include or represent identified sitewide dependencies will be addressed in a separate ISR task (e.g., constructing multi-unit Level 1 risk model) with inputs from analysts (including, for example, Level 1 PRA risk insights or risk-importance measures and MUPRA experience documented in IAEA [2019, 2021a] and EPRI [2021a]) and with consideration of the current state-of-the-art.

Although MU or sitewide IEs were already addressed in Phase 1 (see Section B.5), it is recognized that analysts can identify other such IEs in the process of identifying Phase 2 (or even Phase 3) dependencies.

For Phase 2, some of the dependencies shown in Table B-1 may already be modeled, or nearly so. For this reason, when the dependencies involving shared or connected SSCs for the reactors are identified, the analyst should think ahead to how such dependencies will be represented in the MUPRA such that:

- Basic event (BE) names for SSCs in the Unit 1 model that are not shared with Unit 2 are named such that the parallel BEs in the Unit 2 model can be appropriately named.

- BEs for shared SSCs are uniquely named so that these SSCs are reflected as serving both units.

- Cross-ties or interconnections between the reactor units are appropriately represented.

B.6.1 Shared Physical Resources

Shared resources (e.g., electric power via common grid and/or switchyard, ultimate heat sink, water supplies for fire protection, and diesel fuel) should be identified in the sitewide assessment. Such shared resources may be modeled already in the PRA via support systems. Other shared resources (e.g., some water or diesel fuel supplies) may not be considered in Level 1 PRAs directly but may be implied by modeling in Level 2 PRAs or in modeling FLEX strategies. Note that adequate staffing is NOT considered a physical resource in the L3PRA project categorization of sitewide dependencies. However, if concerns about adequate staffing occur to the analyst while performing the assessment of shared physical resources, notes can be made during this assessment then later transferred to the documentation of “human or organizational dependencies.”

The worksheet shown in Table B-5 (with illustrative examples shown in red font text) was developed to help the analyst identify any of these potential dependencies related to shared physical resources between the two reactor units for:

- all the identified MUIEs
- (as resources allow) the following prioritized list of initiating events: [50]
    - internal events (other than the identified MUIEs) and floods
    - seismic events (other than identified MUIEs)
    - internal fires
    - other external hazards
- FLEX strategies
- Level 2 PRAs for internal events, internal floods, internal fires, seismic, winds, and other hazards

Similarly, the worksheet shown in Table B-6 (with illustrative examples shown in red font text) was developed for analysts to identify dependencies between the SFPs and/or DCS with the two reactor units. However, only identified MUIEs for the two reactors need to be considered for assessment of dependencies between the SFPs and/or DCS with the reactors. Both tables show the identified shared physical resources in the first column, then the associated, relevant IEs and hazards in the third and fourth columns, respectively. Example entries are shown in italicized red font. Inputs for Level 1 PRAs, FLEX strategies, and Level 2 PRAs are accommodated in separate rows.

[50] Unless the identification of MUIEs suggests that a different ordering should be used.

Table B-5 Shared Physical Resources Between the Two Reactor Units

Identified Dependencies	Relevant IEs and MUIEs	Relevant Hazards	Modeling or Screening Notes
Level 1 PRAs			
*Electric power – “extra source”**	*LOOPs*	*Internal events, seismic events, etc.*	*If included in MUPRA, Unit 1 can be credited, but not Unit 2.*
Ultimate heat sink & intake structure	*LONSCW*		*MUIE.*
Water supplies for fire protection	*Any fire IE*	*Internal fires*	
FLEX Strategies			
Diesel fuel needed for FLEX diesel generators and FLEX pumps for Units 1 and 2	*LOOPs*	*Internal events, seismic events*	*FLEX Implementation Plan should address the adequacy of diesel fuel supplies for Phase 2 response. Each unit has its own FLEX DG and pump.*
Refueling trucks?	*LOOPs*		*How many refueling trucks are there?*
Level 2 PRAs			
Diesel fuel needed for B.5.b pumps for Units 1 and 2			*Two B.5.b pumps – one for each unit?*
*To avoid making this document “proprietary,” the specific name of the resource has not been used.			

Table B-6 Shared Physical Resources Between the SFPs and DCS with the Reactors

Identified Dependencies	Rx MUIEs	Relevant Hazards	Modeling or Screening Notes
Level 1 PRAs			
Electric power – grid, switchyard, & weather-related	*LOOPs*	*Internal events, seismic events*	*MUIEs and sitewide IE for SFPs and DCS.*
Ultimate heat sink & intake structure	*LONSCW*		*Is it an MUIE? Is it needed for SFPs?*
FLEX Strategies			
Diesel fuel needed for FLEX pumps to inject water; same FLEX pumps as for Units 1 and 2	*LOOPs*	*Internal events, seismic events*	*SPFs*
Water supply needed for FLEX pumps to inject water; same FLEX pumps as for Units 1 and 2			*SFPs*
Level 2 PRAs			
Diesel fuel needed for b.5.B pumps to inject water; same b.5.B pumps as for Units 1 and 2			*SFPs*
Water supply needed for FLEX pumps to inject water; same FLEX pumps as for Units 1 and 2			*SFPs*

In some cases, for a limitation that is identified and represented in the MU risk model (e.g., an electric power source that can supply only one reactor), a prioritization scheme can be developed, then represented explicitly in MU risk models. Preferably, such a prioritization scheme would be based on plant-specific, formal procedures (e.g., EOPs, FLEX response guidelines, or SAMGs) rather than analyst judgment or operator interviews conducted early in the L3PRA project. However, in the absence of such definitive information, credible assumptions can be made (e.g., Units 1 and 2 use all supplies needed for accident response, and the SFPs are modeled as getting none of these resources). Another approach would be to determine (or assume) the timing of resource needs, then allocate the supplies based on the expected timing of these needs. In all such cases, once the information is known or an assumption is made, explicit changes can be made to the risk models. Decisions such as these would be made when the MU risk models are constructed.

Examples of shared physical resources are given below for different PRA types and hazards.

B.6.1.1 Shared Physical Resources Identified from Level 1 PRAs

Electric power sources (i.e., grid and switchyard) are likely shared across the site across all radiological sources. However, losses of electric power should be considered as part of sitewide or multi-unit initiating events analysis. A similar approach should be taken with common ultimate heat sinks, cooling water intakes structures, etc.

Other shared physical resources (e.g., water supplies for fire protection, diesel fuel, and instrument air) should be identified and then considered as potential dependencies if such supplies, when needed for a sitewide event, could be considered limited. Such limitations might be relevant for only certain initiating events.

B.6.1.2 Shared Physical Resources Identified from FLEX Strategies

As for the Level 1 PRAs, shared resources should be identified for FLEX strategies. Those resources that are directly addressed by the FLEX Implementation Plan (FIP) should be noted as such. Any resources that are not addressed by the FIP would be of particular interest.

B.6.1.3 Shared Physical Resources Identified from Level 2 PRAs

As for the Level 1 PRAs, shared resources should be identified for Level 2 PRAs. Like the Level 1 PRAs, it is expected that most shared resources would be directly modeled in the PRA (e.g., air supplies for containment vent valves). However, some resources may not be directly modeled (e.g., diesel fuel for B.5.b pumps or water supplies from fire protection tanks).

B.6.2 Shared or Connected SSCs

Table B-7 and Table B-8 (with illustrative examples shown in red font text) were developed to document dependencies due to shared or connected SSCs for the two reactors and dependencies between the SFPs/DCS and the two reactors, respectively.

EPRI (2021a) provides guidance regarding the identification and modeling of shared or connected systems and components. In some cases, the single unit PRA (SUPRA) may already include some of this modeling. However, while shared systems and components may be credited in the SUPRA, only one reactor unit can credit a shared system or component in the

MUPRA. While the internal events Level 1 PRA may be the source of most shared or connected systems and components, it is important to document which such systems and components are important to the results for other hazards and for the Level 2 PRA.

Shared or connected structures may have been identified in internal flood and internal fire PRAs. The Level 2 PRAs also may have considered the availability and/or accessibility of equipment (and associated operator actions) for some locations inside the plant. In general, shared or connected structures should be addressed within the appropriate hazard group (e.g., internal fire or flood) and/or PRA level (e.g., Level 1 or Level 2).

B.6.3 Phase 2 Guidance for Assessing “Loosely Coupled” or “Tightly Coupled” Reactor Units

EPRI (2021a) states that the extent of “coupling” between two reactor units is one way to characterize MU risk. Each analyst was asked to consider how they would evaluate the coupling between the units on the reference site. The final determination of coupling was made based on the input of all team members.

Examples of features for “loosely coupled” units that EPRI (2021a) provides are:

- limited (or no) shared systems
- major structures are separated and/or unconnected

In contrast, EPRI (2021a) defines “tightly coupled” units as having complex dependencies, including:

- shared support systems
- shared front-line systems
- inter-unit electrical dependencies
- common or shared structures

EPRI (2021a) states that both “loosely coupled” and “tightly coupled” units have the following types of dependencies:

- shared or common offsite power connections
- shared ultimate heat sink or cooling source
- common component types
- shared accident resources (e.g., FLEX equipment)
- common physical location
- common EOPs, operator training, etc.
- common emergency operations center

Table B-7 Shared or Connected SSCs Between the Two Reactors

Category	Identified Dependencies	Relevant Hazards and MUIEs	Modeling and Screening Notes
Level 1 PRAs			
Shared or connected systems and components	*Nuclear service cooling water system and intake structure*		
Shared or connected structures	*Units 1 and 2 share/have connected MCRs and control buildings*	*Internal fires*	*Shared MCR is a special case that should be addressed by HRA. In some fire scenarios, both MCRs may need to be abandoned due to uninhabitability.*
	Units 1 and 2 share the turbine building	*Internal fires, internal floods, seismic events*	*Other shared or connected structures should be treated in a manner consistent with the hazard group (e.g., internal floods or fires, seismic event) that addresses the structures.*
	Units 1 and 2 have adjacent or connected auxiliary buildings	*Seismic events*	
FLEX Strategies			
Shared or connected systems and components			
Shared or connected structures	*FLEX building*		*One FLEX building. However, the FLEX building has been designed to survive external hazards (e.g., high wind, external floods, and seismic events).*
Level 2 PRAs			
Shared or connected systems and components			
Shared or connected structures			

Table B-8 Shared or Connected SSCs Between the SFPs and DCS with the Reactors

Category	Identified Dependencies	Relevant Hazards and MUIEs	Modeling and Screening Notes
Level 1 PRAs			
Shared or connected systems and components	*SFPs are connected to each reactor via the refueling water canal.*	*Loss of inventory during reactor shutdown initiating event?*	*This scenario also involves a human error as part of its description. Though reactor low power and shutdown is out of scope, sensitivity studies or additional calculations would be prudent. Very important for SFP risk on its own.*
Shared or connected structures	*Are SFPs connected to buildings for Units 1 and 2?*		
FLEX Strategies			
Shared or connected systems and components	*Are the FLEX pumps needed for Units 1 and 2 also the same FLEX pumps needed for SFP injection?*		
Shared or connected structures	*FLEX building*		
Level 2 PRAs			
Shared or connected systems and components	*Are the B.5.b pumps needed for Units 1 and 2 also needed for SFP injection?*		
Shared or connected structures			

B.7 Guidance for Performing Phase 3 Sitewide Dependency Assessments

The scope of the Phase 3 assessment of potential sitewide dependencies consists of the remaining categories of dependencies shown in Table B-1. However, the actual implementation of the Phase 3 assessment can apply to a reduced set of dependency categories if there is loose coupling between the reactor units. The potential sitewide dependency categories assessed in Phase 3 are:

- identical components
- proximal dependencies (relevant mostly for external events)
- human and organizational dependencies
- potential accident propagation between units (may not need to be considered for loosely coupled reactors, especially if there are no shared support systems)
- potential hazards correlations (relevant for seismic events, especially)

As stated in Section 4.3.3, potential sitewide dependencies that are identified in Phase 2 are expected to be more important and are most likely represented with modifications to logic models. On the other hand, potential dependencies identified in Phase 3 are:

- typically modeled by adjustments to BE probabilities, rather than logic modeling
- difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether CCF groups should be expanded and what adjustment factor to use for an expanded group)
- difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the technical support center (TSC), etc.
- typically require modeling that is beyond the PRA state-of-the-art

As in the steps to identify sitewide IEs, other (Phase 3) dependencies between the two reactors are identified first. Dependencies between the SFPs and two reactors are identified next, then dependencies between DCS and the two reactors.

As was stated previously, some of these types of dependencies can be difficult to assess (due to limitations in the availability of information or the state-of-the-art for PRA or hazard modeling). Therefore, it is recommended that analysts view the identification of such dependencies in a way similar to that for the identification of sources of uncertainty. In other words, the analyst should give his/her best effort to identifying such dependencies while recognizing that not all (and, maybe, only a few) of such dependencies can be represented in risk models. Also, the analyst should not be deterred from identifying a potential dependency if such a dependency is beyond the state-of-the-art for PRA or hazard modeling. The decision to include or represent identified sitewide dependencies is addressed in a separate ISR task (e.g., construct multi-unit Level 1 risk model) with inputs from analysts (including, for example, Level 1 PRA risk insights or risk-importance measures, MUPRA experience documented in EPRI [2021a] and IAEA [2019, 2021a]) and with consideration of the current state-of-the-art.

To conserve resources, Phase 2 should be performed first to determine whether the reactors on the reference site can be treated as *loosely* or *tightly* coupled. If the reactors can be treated as *loosely* coupled, then the Phase 3 assessment can be reduced (as discussed further below).

B.7.1 Guidance for Identifying Identical Components and Modeling Cross-Unit CCFs

Worksheets such as those shown in Table B-9 and Table B-10 (with illustrative examples shown in red font text) were used to document the identification of identical components between the two reactors, then between the SFPs/DCS and the two reactors, respectively. The tables and associated approach used for this identification is similar to that used for shared physical resources (Section B.6.1) and shared or connected SSCs (Section B.6.2).

For the reactors, it was expected that the majority of CCFs relevant to MU risk would be identified from the single unit, Level 1 internal events PRA model. However, it was recognized that additional CCF groups could be modeled in Level 1 PRAs for other hazards or in the Level 2 PRAs. Consequently, analysts were asked to document any risk significant CCF groups for other hazards (e.g., fire and seismic) even if these groups had already been identified from the Level 1 internal events PRA. Such information could be helpful in later screening for the ISR task.

Identical components that are modeled in both reactor units were considered for modeling cross-unit CCFs. Two different types of CCFs were identified and documented for potential consideration in MU risk:

- CCFs that are already included in the L3PRA project PRA models
- new CCFs involving identical components across the two reactor units (or between the reactors and the SFPs or DCS)

Note that this identification did not include CCFs for any components not already included in the existing L3PRA project PRA models (which is a scope choice made for the L3PRA project).

For CCFs already modeled in the single unit base PRA, there can be two cases: (1) the existing CCF group size is also appropriate for a multi-unit risk model, or (2) the existing CCF group size has to be expanded for the multi-unit risk model. For example, if there are CCFs already modeled for a system that is shared by both reactor units (e.g., CCFs of service water pumps)[51] and the success criteria is not changed when going from the single unit to multi-unit model, then no expansion of the common cause component group (CCCG) is needed. However, if the CCFs in the single unit PRA model are not in a shared system (e.g., CCF of emergency diesel generators [EDGs]), the CCCG would need to be expanded to address the components in both reactor units and new CCF parameters would need to be estimated.[52]

[51] Note, this is not the case for the service water pumps at the reference plant.

[52] For example, see two NRC sources for how to make such estimations: (1) an INL link to "CCF Parameter Estimations: 2015 Update" for SPAR models, and (2) a 1998 report on CCF parameter estimation (NUREG/CR-5485).

Also, there could be a need for new CCF groups representing any single component in individual reactor models (e.g., the turbine-driven auxiliary feedwater pump) that would be important in a MUPRA.

Screening decisions on which CCFs are modeled in the multi-unit risk model were made in a later ISR task step (e.g., it may not be practical to extend all CCF groups to cross-unit CCF groups). However, analysts were asked to document any relevant screening inputs such as low risk significance (e.g., based on importance measures, such as Fussell-Vesely < 0.005 or Risk Achievement Worth < 2) or limited operational experience to support calculations of CCF parameters for large group sizes.

The approach for finding identical components between the reactors and the SFPs and DCS is similar to that described above. For example, the SFP PRA was reviewed to identify any identical components with the two reactors. Since there are no parallel systems between the SFPs and the reactors, commonalities in equipment design and function were used as the basis for this review.

Table B-9 Identical Components Between the Two Reactors

Identified CCF Groups	Original Group Size	Expanded Group Size	Relevant Hazards and MUIEs	Modeling and Screening Notes
Level 1 PRAs				
EDGs in both Unit 1 and Unit 2	*2*	*4*	*LOOPs*	*Expand CCF group to address potential inter-unit CCFs.*
Turbine-driven auxiliary feedwater pumps for Units 1 and 2	*1*	*2*		*Address as a new CCF group that is cross-unit.*
Nuclear service water pumps				*Is group size in single unit PRA sufficient for MUPRA or do success criteria (and CCF group size) need to be adjusted for MUPRA?*
FLEX Strategies				
FLEX diesel generators				*Already addressed in FLEX PRA?*
FLEX pumps				*Already addressed in FLEX PRA?*
Level 2 PRAs				
b.5.B pumps				

Table B-10 Identical Components Between the SFPs/DCS and the Two Reactors

Identified CCF Groups	Original Group Size	Expanded Group Size	Relevant Hazards and MUIEs	Modeling and Screening Notes
Level 1 PRAs				
Motor-operated valves for SFPs?				
Air-operated valves for SFPs?				
FLEX Strategies				
Level 2 PRAs				

B.7.2 Guidance for Identifying Proximity Dependencies

Proximity dependencies arise from:

- exposure of multiple SSCs to shared phenomenological or environmental conditions
- common features between units
- operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source

Proximity dependencies may cause failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat or cold, radiation levels), debris, explosions,[53] etc., from a nearby radiological source. External hazards may fail identical or similar structures due to common location of structures for both units. These dependencies are not likely to have been identified in individual risk models for each radiological source. External hazards and radiological concerns (e.g., high radiation areas identified as part of Level 2 PRA) are likely to be the principal concern for SSCs in both units that share phenomenological or environmental conditions. Dependencies related to common features between units (e.g., structures for both units are in essentially in the same location or structures for both units are the same height) are likely to be important only to external hazards. Proximity dependencies involving operator actions (e.g., field operator actions taken for Unit 2 while near Unit 1) might be identified in this category but should be addressed in the category “human or organizational” dependencies (see Section B.7.3 below).

Proximity dependencies for SSCs due to environmental conditions (that are not associated with external hazards) can only occur if SSCs for both reactors are shared or connected. Although shared or connected SSCs are addressed in the Phase 2 assessment of sitewide dependencies, additional assessment from the perspective of proximity dependencies should be performed. Examples of the types of hazards that involve environmental conditions that could affect SSCs due to proximity include:

- effects of fire events (e.g., heat, smoke, toxic gases)
- radiation (for both Level 1 PRA and Level 2 PRA conditions)
- internal flooding

By the definition given above, proximity dependencies related to external hazards are similar or overlap those for the hazard correlation category of potential sitewide dependencies.

B.7.3 Guidance for Identifying Potential Human or Organizational Dependencies

This category has been defined differently in the EPRI report (EPRI, 2021a) and the two IAEA reports (IAEA 2019, 2021a). The definition used in the L3PRA project ISR approach is intended to capture all remaining dependencies related to human and organization resources. It is also expected that some potential dependencies identified in other categories (e.g., shared physical resources or proximity dependencies) are most appropriately addressed by human reliability analysis (HRA).

[53] The L3PRA project has not produced any results that include the potential for explosions.

As indicated in Table B-1, the definition for the “Human or Organizational” category of dependencies is:

> Dependencies between operator actions across multiple radiological sources that can result from multiple causes, including sharing of staff and shared organizational factors.

The following are examples of potential human and/or organizational dependencies discussed in EPRI (2021a) and IAEA (2019, 2021a):

- shared human resources between units
- shared control rooms
- common procedures (e.g., EOPs, AOPs, SAMGs, FLEX procedures)
- common operator training
- common human machine interface (HMI)
- common command and control structure (C&C)
- common TSC
- common emergency response organization (ERO)
- common offsite support
- increased stress due to MU accident conditions
- accessibility concerns due to the other unit’s degraded condition
- common environmental concerns for operators of both units (e.g., field operators taking actions at local control stations, at locations shared by both units, or outside the plant[s])

In addition, typical HRA concerns are relevant, such as:

- timing of the action (especially with respect to when conditions from one reactor can affect another reactor)
- cues and indications to prompt and/or support operator actions
- potential dependencies with prior actions

B.7.3.1 Basis for Modeling Sitewide Human or Organizational Dependencies

It should be noted that there is limited information on how sitewide human or organizational dependencies are modeled in MUPRAs beyond the treatment of operator actions related to shared human resources or shared/connected SSCs. In addition, addressing some of these dependencies has been recognized as being beyond the current state-of-the-art, similar to that for treatment of cross-unit CCFs.

EPRI (2021a) refers to dependencies arising from shared physical resources and shared or common SSCs as “explicit” dependencies. All other potential dependencies are referred to as “implicit” or “indirect” dependencies (e.g., features of shared plant contexts, such as a shared control room and TSC and common procedures and training). “Implicit” dependencies can be addressed using modeling assumptions or subjective judgment to adjust human error probabilities (HEPs).

Note that the “explicit” human and organizational dependencies are tied to potential dependencies identified in Phase 2 (i.e., physical resources and shared or connected SSCs). In addition, the MUPRA pilot studies in EPRI (2021a) seem to have focused on dependencies that change the feasibility of operator actions, in particular, “explicit” dependencies. Other potential human and organizational dependencies that could be considered “explicit” and result from changes in feasibility are those operator actions that are affected by environmental conditions from the other unit (i.e., proximity dependencies).

In addition to the EPRI terminology for “explicit” and “implicit/indirect” human and organizational dependencies, the current guidance also adopts EPRI’s expectation that “explicit” human and organizational dependencies are more important than “implicit/indirect” dependencies.

B.7.3.2 Identifying “Explicit” Sitewide Human or Organizational Dependencies

Within the definition of “explicit” potential human or organizational dependencies, there are two types of potential dependencies to identify:

- dependencies that are directly tied to shared physical resources and shared or connected SSCs
- dependencies that result in an operator action being no longer feasible

The approaches for identifying these “explicit” dependencies are discussed below.

<u>Identifying Potential Human and Organizational Dependencies Associated with Shared Resources and Shared or Connected Systems and Components</u>

It is expected that the most important human and organizational dependencies are those associated with the Phase 2 potential sitewide dependencies between the two reactors, and the SFPs with the two reactors (i.e., shared physical resources and shared or connected SSCs). (See, for example, the discussion in Section 6.2.2 in EPRI [2021a]).

The recommended steps for identifying these potential human and organizational dependences are:

Step 1: Review Phase 2 potential sitewide dependencies for shared physical resources that require an operator action in order to be used.

Step 2: Review Phase 2 potential sitewide dependencies for shared or connected SSCs that require an operator action to use.

Step 3: Review (as needed) Level 1 PRA single source cutsets for operator actions for potential dependencies that have not been otherwise identified.

Step 4: Review (as needed) Level 2 PRA cutsets for all radiological sources to identify any other potential dependencies.

Step 5: Review multi-unit results for multiple operator actions in a single cutset.

In the first two steps, after the Phase 2 potential sitewide dependency results have been reviewed, the PRA lead should be consulted to help identify the potential sitewide operator dependencies and to provide any additional needed information on these operator actions. It should be noted that potential dependencies found in Step 1 might actually result in operator actions becoming infeasible (because physical resources are not adequate to support multiple radiological sources) (see further discussion below). Steps 1 and 2 should be performed in advance of developing any sitewide or multi-unit models.

Step 3 is performed just before the multi-unit model is developed as a final check on what dependencies should be considered in the Level 1 MUPRA model. Similarly, Step 4 is a check of the Level 2 PRA cutsets for the reactors as well as the SFPs and DCS.

Step 5 is similar to traditional HRA/PRA dependency analysis in the sense that this review is performed after the multi-unit and sitewide models have been developed and initial results are produced. Consequently, this review is performed much later in the overall ISR task.

Identifying Potential Human and Organizational Dependencies That Result in Infeasible Operator Actions

Operator action feasibility criteria were developed specifically for the fire PRA context (see NUREG-1921 [NRC, 2012b] and its Supplement 1 [NRC, 2017a] and Supplement 2 [NRC, 2019]) but are applicable to other PRA hazards, including radiological concerns for Level 2 PRA HRA. These feasibility criteria also have been adopted by the L3PRA project, as needed (e.g., for Level 2 PRA HRA).

The feasibility factors for an operator action from the fire HRA reports (NRC, 2012b, 2017, 2019) are:

- sufficient time available for the action
- sufficient staffing for the action
- primary cues available and sufficient for the action
- action is proceduralized and trained upon
- action location (and travel paths) are accessible (including consideration of environmental factors)
- needed equipment and tools are available and accessible
- relevant components are operable
- action is supported by a communications plan
- action is supported by a plan for command and control

If any one of the above statements are not true, then the operator action should be considered infeasible.

For the purposes of this assessment of potential dependencies, the criteria that are not likely to have been addressed as part of the evaluation of other potential sources of dependencies are:

- time available (and, maybe time required, that is changed due to operator actions or conditions for other radiological sources)
- sufficient staffing
- action location and travel path accessible (due to shared or connected structures, in particular)
- needed equipment available and accessible

So, the analyst should consider if these specific feasibility criteria would be assessed differently (i.e., no longer feasible) when considering a multi-unit and sitewide (i.e., including actions for the SFPs) accident. In addition, some assumptions may need to be made about these factors (then re-visited as the analysis proceeds or for sensitivity analyses).

Two main steps for identifying dependencies that result in an operator action being no longer feasible are:

Step 1: Identify any changes to the feasibility for local operator actions that are taken for the reactors (e.g., a local action for Unit 2 reactor that might no longer be feasible because of either conditions for Unit 1 or operator actions for Unit 1).

Step 2: Identify any local operator actions for the SFPs (or DCS) that might no longer be feasible because of conditions for either (or both) of the reactors or because of operator actions taken for either (or both) of the reactors.

Also, while some such dependencies might be anticipated before the multi-unit or sitewide risk models are developed, these steps should be re-visited when the models are developed, and their results reviewed.

Examples of such dependencies (for which the human error probability, or another type of BE probability, would need to be changed to 1.0) are:

- A fire in Unit 1 main control room (MCR) generates enough smoke in both MCRs to satisfy the criteria for MCR abandonment (see the fire HRA reports [NRC, 2012b, 2017, 2019]) of both units (i.e., shared structure with a common environmental hazard).
- Water resources needed to implement extensive damage mitigation guideline (EDMG) strategies for both reactors are only sufficient for one reactor.
- There are not enough field operators to simultaneously implement EDMG strategies for both reactors and the SFPs.

- Unit 1 reaches core damage before Unit 2 and the resulting high radiation from Unit 1 prevents the performance of a local operator action for Unit 2 (i.e., proximity of environmental hazard from Unit 1 for operator actions needed for Unit 2).

B.7.3.3 Identifying "Implicit/Indirect" Sitewide Human or Organizational Dependencies

At the beginning of Section B.7.3, the following list of example human and organizational dependencies was given, most of which represent potential implicit/indirect dependencies:

- shared human resources between units
- shared control rooms
- common procedures (e.g., EOPs, AOPs, SAMGs, FLEX procedures)
- common operator training
- common HMI
- common C&C
- common TSC
- common ERO
- common offsite support
- increased stress due to MU accident conditions
- accessibility concerns due to the other unit's degraded condition
- common environmental concerns for operators of both units (e.g., field operators taking actions at local control stations, at locations shared by both units, or outside the plant[s])

Two of these factors have already been addressed (at least, in part) in Section B.7.3.2, specifically:

- The first item corresponds with the feasibility criterion for sufficient staff and accessibility – both action location and travel path.
- The second-to-last last item corresponds with the feasibility criterion for accessibility, such as common environmental hazards (e.g., smoke and heat from fires, debris from seismic events, debris and water from external flooding events, or radiation for Level 2 PRA) for operator actions in both units.

The worksheet shown in Table B-11 provides a means for documenting information regarding the remaining potential sitewide human and organizational dependencies. Example entries with the reference site in mind have been included in this table. Also, discussion from Section 6.5 in EPRI (2021a) informed the examples of potential positive and/or negative impacts of these dependencies, and notes for potential modeling. As noted above, modeling these dependencies is beyond the current state-of-practice, but it is possible that compelling information exists to support such modeling. How to treat such potential dependencies is a candidate for future research.

Table B-11 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies

Characteristic of Potential Dependency	Y/N	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Shared MCR	N	Operators distracted by alarms on other unit; "group think" that is incorrect; loss of "swing" operator	Face-to-face communication; "group think" that is correct; sharing "swing" operator; closer coordination between units	EPRI (2021a) did an initial comparison between shared and connected MCRs and preliminarily did not find any significant differences between the two.
Connected MCR	Y	Potential distractions for operations managers; "group think" that is incorrect	Face-to-face communication; "group think" that is correct; easier coordination between units	See above
Common procedures	Y	If there is a weakness, it likely will affect actions for both units.	If the procedural support is good, actions should be independent.	Weaknesses or "gaps" might be considered for explicit modeling (e.g., if action for one unit is failed due to "gap," then action for second unit also considered to be "failed").
Common training	Y	Same as for "procedures"	Same as for "procedures"	Same as for "procedures"
Common HMI	Y			
Common C&C	Y[a]	Same as "connected MCR"; challenge of responding to multiple reactors within the same time period.	Same as "connected MCR"	EPRI (2021a) suggests that on-site C&C should be a net positive.
Common TSC	Y[b]			
Common ERO	Y			
Common offsite support	Y			
Increased stress due to MU accident	?			???

B.7.4 Guidance for Identifying Potential Accident Propagation Between Radiological Sources

The principal guidance for this Phase 3 category of sitewide dependency assessment is that the results of the Phase 2 assessment of coupling between the two reactors should be performed first. If the Phase 2 assessment is that the reactors are “loosely coupled,” then this assessment of accident propagation can be limited.

At present, there is little guidance on the identification of potential accidents propagating from one unit to another. The IAEA (IAEA, 2019; IAEA, 2021a) and EPRI (EPRI, 2021a) reports identify this category of sitewide dependency but provide little or no additional information. In practice, the only types of initiators that have been identified as potentially propagating from one unit to another are fires (which are already modeled in Level 1 fire PRAs). Note that in Section 8, the potential for hydrogen combustion affecting the SFPs was identified as a potential dependency for Level 2 PRA. Note that this analysis has not determined whether hydrogen combustion in the reactor containment should be categorized as a proximity failure or a propagating event. However, as noted earlier, the ultimate goal is to identify potential dependencies regardless of how they are labeled.

B.7.5 Guidance for Identifying Potential Sitewide Hazards Correlations

This category of dependencies addresses SSCs and operator actions that may be affected in the same or similar ways. For example, potential hazards correlations are especially relevant for seismic events (but may also be relevant for other external hazards). Such dependencies are likely to be already addressed in the base PRA for each unit. However, such coincident failures of operator actions and/or SSCs for each radiological source need to be accounted for in the overall site model.

The analyst should use judgment and knowledge of the current state-of-the-art to select and represent multi-unit and sitewide hazard correlations. Examples of papers consulted by the project team include Abrahamson (1993), Kawakami (2003), Zerva (2009), and DeJesus-Segarra (2020).

APPENDIX C
IDENTIFICATION AND ANALYSIS OF MULTI-UNIT AND SITEWIDE INITIATING EVENTS

This appendix presents the results for the Phase 1 sitewide dependency assessment as part of the integrated site risk (ISR) task. In addition, this appendix discusses the selection of multi-unit initiating events (MUIEs) or sitewide initiating events (IEs) to represent in the ISR task. Also, the calculation and selection of sitewide IE frequencies is presented. Some of this material also appears in Section 5.

C.1 Approach for Phase 1 Sitewide Dependency Assessment

Section B.5 describes the approach used for the Phase 1 sitewide dependency assessment. The principal basis for identifying both multi-unit initiating events (MUIEs) and sitewide initiating events (IEs) for spent fuel pools (SFPs) and dry cask storage (DCS) is the converse of the screening criteria given in Section B.5. If any of these converse criteria are met, then the potential MUIE or sitewide IE was retained for consideration in the Level 3 PRA (L3PRA) project's ISR assessment:

1. The IE immediately results in reactor trip in both units.

2. The IE immediately results in reactor trip of one unit and a degraded condition in the second unit.

3. The IE immediately results in degraded conditions in both reactor units.

Note, in the above criteria, "degraded condition" is defined to include the eventual occurrence of a reactor trip, either an automatic trip or a required manual trip.

In the approach described in Appendix B, individual analysts who were most knowledgeable of the various L3PRA models (e.g., different PRA hazards and types) reviewed the respective PRAs and associated materials. Consistent with other published guidance (e.g., IAEA, 2019; EPRI, 2021a), lists of IEs used in Level 1 PRAs, and response to such IEs, are the focus of the Phase 1 sitewide dependency assessments. Assessments for the reactors were performed first. Then, assessments for the SFPs and DCS were performed (since the SFP and DCS PRA results indicate a smaller set of relevant IEs than for the reactors). The assessments for the reactors were performed in succession, each analyst building on the previous assessment. The order of inputted results from analysts was:

1. internal events Level 1 PRA for the two reactors
2. internal floods Level 1 PRA for the two reactors
3. fire, seismic, and wind Level 1 PRAs for the two reactors
4. all hazards and Level 1 PRAs for the SFPs[54] and DCS

A group brainstorming and discussion session followed the individual assessments to confirm, expand, or refine the individual results, as needed.

[54] Note that the two spent fuel pools are treated as a single large pool in the L3PRA project because they are hydraulically connected for most plant operating states.

C.2 Results for Phase 1 Sitewide Dependency Assessment

Results for the identification of MUIEs or sitewide IEs by hazard for both the two reactors and the SFPs and DCS are given in Section C.2.1 and Section C.2.2, respectively. Section C.2.3 provides a summary of all these results.

C.2.1 Results for the Identification of MUIEs for the Two Reactors

The results for the identification of potential sitewide IEs for the reactors are given for internal events in Section C.2.1.1, for internal floods in Section C.2.1.2, and for fire, high winds, and seismic events in Section C.2.1.3.

The guidance in Section B.5 recommends the use of two different tables for the documentation of potential MUIEs. In particular, different tables were used to document:

- IEs that satisfy the converse screening criteria (see Table B-2)
- IEs that have been screened out (see Table B-3)

C.2.1.1 Reactors: MUIEs for Internal Events

Table C-1 presents the results of the identification of potential MUIEs using the internal events Level 1 PRA (NRC, 2022d) and associated analyst knowledge and understanding. This table combines the fields and documentation provided by Table B-2 and Table B-3 in Appendix B. In other words, it shows which IEs satisfy the converse screening criteria as well as those IEs that do not satisfy the criteria.

The results given in Table C-1 can be summarized as follows:

- Losses of offsite power (LOOPs) should be assessed as MUIEs if they are:
    - grid-related
    - switchyard-centered
    - weather-related
- Three other potential MUIEs are recommended to be not screened out, although the criteria for screening are met:
    - loss of nuclear service cooling water (NSCW)
    - interfacing system LOCA (ISLOCA) from residual heat removal (RHR) hot leg suction lines
    - ISLOCA from RHR cold leg injection lines [two IEs]
- All other IEs considered in the internal events Level 1 PRA are screened out.

For the three IEs that should be assessed as MUIEs, the converse criterion met is:

- The IE immediately results in reactor trip in both units.

Table C-1 Level 1 PRA Internal Events Screening for MUIEs

Potential MUIE [# of Initiating Events per Unit]	Converse Criteria Met?	Refinement/Caveat Notes	Internal Events CDF (%)	Screened Out?	
				Why?	Potential Negative Consequences?
Grid-related loss of offsite power (LOOP)	Yes (#1)	Definite sitewide LOOP would occur.	1.8E-5 (29%)		
Switchyard-centered LOOP	Yes (#1)	Could result in sitewide or single unit LOOP.	1.0E-5 (16%)		
Weather-related LOOP	Yes (#1)	Likely sitewide LOOP, but not definite.	9.0E-6 (14%)		
Loss of nuclear service cooling water (NSCW)	No	If cross-unit common-cause failure (CCF) is considered, dual-unit loss of NSCW can occur. The dominant loss of NSCW cutsets are from pump CCFs.	8.8E-6 (14%)	This scenario is not recommended to be screened out. Due to the risk significance of the IE and the dominant CCF aspects, it should be considered as a potential MUIE. If considered, large uncertainties will be associated with CCF parameters due to large common cause component groups (CCCGs) (e.g., 12 pumps and 16 fans). Note that installation of new reactor coolant pump (RCP) seals significantly reduced the risk of this scenario in the FLEX sensitivity case.	
Other transients	No		2.5E-6 (4%)	Should not affect the other unit.	None
Medium loss-of-coolant accident (LOCA)	No	CCF of passive components is not considered. It is not believed that a LOCA on one unit will affect the other.	2.3E-6 (4%)	Should not affect the other unit.	None
Loss of 4.16-kV alternating current (AC) bus [two IEs]	No	Intersystem bus CCFs are not considered in the SUPRA; therefore, consideration of multi-unit CCFs may not be practical.	2.3E-6 (4%)	Evaluation of potential cross-unit CCF is likely not practical. In addition, multi-unit risk of these transients is not expected to be very risk significant unless cross-unit CCF is evaluated and is a significant contributor.	Low
Plant-centered LOOP	No		1.9E-6 (3%)	Should not affect the other unit.	None
Secondary-side breaks [two IEs]	No	CCF of passive components is not considered.	1.7E-6 (3%)	Should not affect the other unit.	None

Table C-1 Level 1 PRA Internal Events Screening for MUIEs (cont.)

Potential MUIE [# of Initiating Events per Unit]	Converse Criteria Met?	Refinement/Caveat Notes	Internal Events CDF (%)	Screened Out?	
				Why?	Potential Negative Consequences?
Loss of 125V direct current bus [two IEs]	No	Intersystem bus CCFs are not considered in the SUPRA; therefore, consideration of multi-unit CCFs may not be practical.	1.2E-6 (2%)	Evaluation of potential cross-unit CCF is likely not practical. In addition, multi-unit risk of these transients is not expected to be very risk significant unless cross-unit CCF is evaluated and is a significant contributor.	Low
Turbine trip	No	[redacted]	1.1E-6 (2%)	Should not affect the other unit.	None
Loss of RCP seal injection	No	If a loss of both units' normal charging pumps (NCPs) due to CCF is considered, then a dual-unit loss of RCP seal injection can occur.	1.0E-6 (2%)	Although a loss of both unit's NCPs could occur via CCF, it is expected to be a low-risk event at both units. In addition, the installation of the shutdown RCP seals significantly reduced this risk of this scenario in the FLEX sensitivity case.	None
Reactor trip	No	[redacted]	9.8E-7 (2%)	Should not affect the other unit.	None
Loss of main feedwater (MFW)	No	If cross-unit CCF is considered (e.g., MFW pumps), dual-unit loss of MFW can occur. A loss of MFW is not substantially more significant than a typical transient for PWRs.	5.2E-7 (<1%)	Although a loss of MFW at both units is possible via CCF, a dual-unit transient of this nature is not expected to be a significant contributor to multi-unit risk.	None
Loss of condenser heat sink (CHS)	No	If cross-unit CCF is considered (e.g., condensate or circulating water pumps), dual-unit loss of CHS can occur. A loss of CHS is not substantially more significant than a typical transient for PWRs.	4.8E-7 (<1%)	Although a loss of CHS at both units is possible via CCF, a dual-unit transient of this nature is not expected to be a significant contributor to multi-unit risk.	None

Table C-1 Level 1 PRA Internal Events Screening for MUIEs (cont.)

Potential MUIE [# of Initiating Events per Unit]	Converse Criteria Met?	Refinement/Caveat Notes	Internal Events CDF (%)	Screened Out?	
				Why?	Potential Negative Consequences?
Loss of auxiliary component cooling water (ACCW)	No	If cross-unit CCF is considered (e.g., ACCW pumps), dual-unit loss of ACCW can occur.	2.5E-7 (<1%)	Although a loss of ACCW at both units could occur via CCF, it is expected to be a low-risk event at both units. Note that the installation of the shutdown RCP seals significantly reduced the risk of this scenario in the FLEX sensitivity case.	None
Small LOCA	No	CCF of passive components is not considered. It is not believed that a LOCA on one unit will affect the other.	2.4E-7 (<1%)	Should not affect the other unit.	None
Interfacing system LOCA (ISLOCA) from residual heat removal (RHR) hot leg suction lines	No	If cross-unit CCF of the RHR hot-leg suction isolation valves is considered, dual-unit ISLOCA can occur. Note that conditional failures were treated through expert elicitation.	2.3E-7 (<1%)	This scenario is not recommended to be screened out. Due to the dominant CCF aspects and high-risk potential of dual-unit ISLOCA, it should be considered as a potential MUIE. If considered, large uncertainties will be associated with CCF of opposite unit MOVs. Expert elicitation may be needed if this scenario is modeled.	
Inadvertent safety injection (SI) actuation	No	[redacted]	1.5E-7 (<1%)	Should not affect the other unit.	None
Steam generator tube rupture (SGTR)	No	CCF of passive components is not considered. It is not believed that an SGTR on one unit will affect the other.	1.2E-7 (<1%)	Should not affect the other unit.	None
Reactor vessel rupture	No	CCF of passive components is not considered. It is not believed that a LOCA on one unit will affect the other.	1.0E-7 (<1%)	Should not affect the other unit.	None

Table C-1 Level 1 PRA Internal Events Screening for MUIEs (cont.)

Potential MUIE [# of Initiating Events per Unit]	Converse Criteria Met?	Refinement/Caveat Notes	Internal Events CDF (%)	Screened Out?	
				Why?	Potential Negative Consequences?
Loss of two 120V AC panels [six IEs]	No	If cross-unit CCF of the 120V AC panels is considered, a loss of two 120V AC panels at each unit can occur. Note that CCF of four panels is not treated in the single unit PRA.	9.6E-8 (<1%)	Although a loss of two 120V AC panels at each unit is possible via CCF, a dual-unit transient of this nature is not expected to be a significant contributor to multi-unit risk.	None
ISLOCA from RHR cold leg injection lines [two IEs]	No	If cross-unit CCF of the RHR cold-leg injection isolation valves is considered, dual-unit ISLOCA can occur. Note that conditional failures were treated through expert elicitation.	8.4E-8 (<1%)	This scenario is not recommended to be screened out. Due to the dominant CCF aspects and high-risk potential of dual-unit ISLOCA, it should be considered as a potential MUIE. If considered, large uncertainties will be associated with CCF of opposite unit MOVs. Expert elicitation may be needed if this scenario is modeled.	
Large LOCA	No	CCF of passive components is not considered. It is not believed that a LOCA on one unit will affect the other.	3.6E-8 (<1%)	Should not affect the other unit.	None
ISLOCA from RCP Stage 1 seal leak-off	No	This event is a conditional event that requires loss of RCP cooling and injection.	3.4E-8 (<1%)	Given the significant risk reduction expected from the installation of the shutdown RCP seals for this scenario, it is recommended that this IE be screened out.	None
Loss of instrument air (IA)	No	If cross-unit CCF is considered (e.g., air compressors), a dual-unit loss of IA can occur. Note that IA is not a safety-related system. A dual-unit loss of IA would result in a loss of CHS at both units.	2.5E-8 (<1%)	Although a loss of IA at both units is possible via CCF, a dual-unit transient of this nature is not expected to be a significant contributor to multi-unit risk.	None

Table C-1 Level 1 PRA Internal Events Screening for MUIEs (cont.)

Potential MUIE [# of Initiating Events per Unit]	Converse Criteria Met?	Refinement/Caveat Notes	Internal Events CDF (%)	Screened Out?	
				Why?	Potential Negative Consequences?
ISLOCA from RCP thermal barrier heat exchanger	No	CCF of passive components is not considered. It is not believed that this type of ISLOCA at one unit will affect the other.	3.5E-11 (<1%)	Should not affect the other unit.	None

For all three potential MUIEs that are recommended to be not screened out (despite meeting the screening criteria), PRA results were dominated by CCFs and cross-unit CCFs could be considered (even if there are large uncertainties in assessing large CCF groups). (Note that this assessment links the Phase 1 and Phase 3 assessments of site dependencies.)

C.2.1.2 Reactors: MUIEs for Internal Floods

Table C-2 presents the results of the identification of potential MUIEs using the internal floods Level 1 PRA (NRC, 2022e) and associated analyst knowledge and understanding. This table is similar to Table C-1 in that it combines the fields and documentation provided by Table B-2 and Table B-3 in Appendix B. In other words, it shows which internal flood IEs satisfy the converse screening criteria as well as those internal flood IEs that do not satisfy the criteria.

The results given in Table C-2 can be summarized as follows:

- Four types of internal floods possibly satisfy the converse criteria:
    - 1-FLI-TB_500_HI1 – Flood in turbine building due to human errors restoring the main condenser after maintenance
    - 1-FLI-TB_500_LF – Flood in turbine building—circulating water (CW) expansion joint failure
    - 1-FLI-TB_500_LF-CDS – Flood in turbine building—condensate system piping failure
    - 1-FLI-TB_500_HI2 – Flood in turbine building due to human errors restoring turbine plant closed cooling water system heat exchangers after maintenance
- All other internal flooding IEs are screened out.

For the four flooding IEs that could be assessed as MUIEs, the converse criterion met is:

- The IE immediately results in degraded conditions in both reactor units.

However, the lead analyst for internal floods PRA recommended screening these IEs out of the MUPRA. In the “Screened Out?” column of Table C-2, the analyst states that, for such flooding events, “[i]mpacts on opposite unit are possible, but unlikely.” Further arguments for screening out each potential MUIE are given in the “Screened Out?” column in Table C-2. Also, it should be noted that the overall core damage frequency (CDF) calculated for the internal flooding, at-power Level 1 PRA is 7.9×10^{-7} per reactor-critical-year (i.e., approximately 1 percent of internal events CDF). Consequently, these “unlikely” internal flooding IEs would be expected to result in very low multi-unit CDFs (MUCDFs).

The lead analyst for the internal flooding PRA provides these additional notes about screening for MUIEs:

- Many of the significant flooding scenarios involve failures of NSCW piping, which impacts the availability of NSCW as well as other equipment impacted by the flood.

However, the plant layout and location of these floods make it unlikely that they would impact both units.

- Turbine building floods have potentially a very large flood source flow rate and volume, which makes them potential candidates for multi-unit impacts. However, the lower level of the turbine building has walls separating the units.

- Main control rooms (MCRs) and train B cable spreading rooms in the control building are identified as areas with potential dual-unit flooding impacts. However, these scenarios were screened from the single unit model (and, therefore, are not addressed in the ISR task). See NRC (2022e), Appendix C, Section C.3.1, for more discussion of the potential dual-unit impacts.

Table C-2 Level 1 PRA Internal Floods Screening for MUIEs

Potential MUIE	Are Any of the Converse Criteria met? (Yes/No) Which?	Refinement/Caveat Notes	PRA Results, Risk Significance, Other Notes	Screened Out?	
				Why?	Potential Negative Consequences of Screening Out IE?
1-FLI-AB_108_SP1	No	Flood in south main steam valve room impacting SG1	7.5% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-AB_108_SP2	No	Flood in south main steam valve room impacting SG4	7.5% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-AB_A20	No	Flood in aux. bldg. rooms A06 and A20	Not significant	Impacts only one reactor unit.	None
1-FLI-AB_C113_LF1	No	Flood in aux. bldg. room C113	19.6% of flooding CDF	Impacts only one reactor unit, NSCW train A	None
1-FLI-CB_122_SP	No	Flood in north main steam valve room impacting SG3	12.9% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-CB_123_SP	No	Flood in north main steam valve room impacting SG2	12.9% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-CB_A48	No	Flood in control bldg. train A 4.16 kvac switchgear room	1.8% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-CB_A60	No	Flood in control bldg. room A60	2.4% of flooding CDF	Impacts only one reactor unit.	None
1-FLI-TB_500_HI1	Yes (possible #3)	Flood in turbine bldg., main condenser post-maintenance error	Not significant	Recommend screening. Impacts on opposite unit are possible, but unlikely. Flood drains to below grade lowest level where there do not appear to be any connections between units. Also, single unit impact is not significant.	

Table C-2 Level 1 PRA Internal Floods Screening for MUIEs (cont.)

Potential MUIE	Are Any of the Converse Criteria met? (Yes/No) Which?	Refinement/Caveat Notes	PRA Results, Risk Significance, Other Notes	Screened Out? Why?	Screened Out? Potential Negative Consequences of Screening Out IE?
1-FLI-TB_500_LF	Yes (possible #3)	Flood in turbine bldg., CW expansion joint failure	2.1% of flooding CDF	Recommend screening. Impacts on opposite unit are possible, but unlikely. Flood drains to below grade lowest level where there do not appear to be any connections between units. Single unit impact is small, but not insignificant.	
1-FLI-AB_B08_LF	No	Flood in aux. bldg. room B08	Not significant	Impacts only one reactor unit, NSCW train A	None
1-FLI-AB_B24_LF2	No	Flood in aux. bldg. room B24	Not significant	Impacts only one reactor unit, NSCW train A	None
1-FLI-AB_B50_JI	No	Flood in aux. bldg. room B50	Not significant	Impacts only one reactor unit, CCW and NSCW train B	None
1-FLI-AB_C115_LF	No	Flood in aux. bldg. room C115	11.7% of flooding CDF	Impacts only one reactor unit, NSCW train A	None
1-FLI-AB_C118_LF	No	Flood in aux. bldg. room C118	Not significant	Impacts only one reactor unit, NSCW train B	None
1-FLI-AB_C120_LF	No	Flood in aux. bldg. room C120	16.5% of flooding CDF	Impacts only one reactor unit, NSCW train A	None
1-FLI-AB_D74_FP	No	Flood in aux. bldg. room D74	Not significant	Impacts only one reactor unit	None
1-FLI-DGB_101_LF	No	Flood in DG bldg. train B	Not significant	Impacts only one reactor unit, NSCW train B	None

Table C-2 Level 1 PRA Internal Floods Screening for MUIEs (cont.)

Potential MUIE	Are Any of the Converse Criteria met? (Yes/No) Which?	Refinement/Caveat Notes	PRA Results, Risk Significance, Other Notes	Screened Out?	
				Why?	Potential Negative Consequences of Screening Out IE?
1-FLI-DGB_103_LF	No	Flood in DG bldg. train A	Not significant	Impacts only one reactor unit, NSCW train A	None
1-FLI-AB_A20_FP	No	Flood in aux. bldg. rooms A20, A11, A12	Not significant	Impacts only one reactor unit	None
1-FLI-AB_D78_FP	No	Flood in aux. bldg. rooms D78	Not significant	Impacts only one reactor unit	None
1-FLI-TB_500_LF-CDS	Yes (possible #3)	Flood in turbine bldg., condensate piping failure	Not significant	Recommend screening. Impacts on opposite unit are possible, but unlikely. Flood drains to below grade lowest level where there do not appear to be any connections between units. Also, single unit impact is not significant.	
1-FLI-TB_500_HI2	Yes (possible #3)	Flood in turbine bldg., turbine plant closed cooling water heat exchanger post-maintenance error	Not significant	Recommend screening. Impacts on opposite unit are possible, but unlikely. Flood drains to below grade lowest level where there do not appear to be any connections between units. Also, single unit impact is not significant.	

C.2.1.3 Reactors: MUIEs for Fires, High Winds, and Seismic Events

Table C-3 presents the results of the identification of potential MUIEs using the seismic, fire, and high wind events Level 1 PRAs (NRC, 2023b; NRC, 2023e; NRC, 2023a) and associated analyst knowledge and understanding. This table is similar to Table B-2 in Appendix B and shows the results of applying the converse screening criteria:

1. The IE immediately results in reactor trip in both units.
2. The IE immediately results in reactor trip of one unit and a degraded condition in the second unit.
3. The IE immediately results in degraded conditions in both reactor units.

The results given in Table C-3 can be summarized as follows:

- Four types of fires satisfy the converse criteria:
    - MCR abandonment scenarios
    - scenarios with shared areas “A+Y”
    - Unit 2 fires that cascade to Unit 1
    - Unit 1 fires that cascade to Unit 2
- All eight bins for seismic events satisfy the converse criteria.
- The high wind scenarios satisfy the converse criteria.
- Other external hazard scenarios are screened out.

Table C-3 Converse Screening Criteria for Including MUIEs in MU Risk: Single Reactor Level 1 PRA Results for Fire, Seismic, and Wind

External Hazard	Scenarios that Are Potential MUIEs	Scenario Description, Characteristics	Are Any of the Converse Criteria Met? (Yes/No) Which?	PRA Results, Risk Significance, Other Notes	Screened Out?	
					Why?	Potential Negative Consequences of Screening Out IE?
Internal Fire Events	MU-IE-FRI-1	Both MCRs evacuated (CCDP = 1); MCR Abandonment	Yes: 1,2	Internal fire events are a major contributor to total single-unit CDF with a 43% contribution. MCR scenarios contribute 14% and MCR abandonment scenarios contribute less than 1% to total internal fire CDF. Others contribute 86% to internal fire CDF and need to be evaluated.		
	MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2; at least MU LOOP				
	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2); at least MU LOOP				
	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1); at least MU LOOP				
Seismic Events	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs	Yes: 1,2	All modeled seismic events are two-unit trips. Seismic events contribute 8% to the total single unit CDF.		
	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs				
	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs				
	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs				
	MU-IE-EQK-5	Seismic event in bin 5 (0.9–1.1g) occurs				
	MU-IE-EQK-6	Seismic event in bin 6 (1.1–1.5g) occurs				
	MU-IE-EQK-7	Seismic event in bin 7 (1.5–2.5g) occurs				
	MU-IE-EQK-8	Seismic event in bin 8 (2.5 and above) occurs				

Table C-3 Converse Screening Criteria for Including MUIEs in MU Risk: Single Reactor Level 1 PRA Results for Fire, Seismic, and Wind (cont.)

External Hazard	Scenarios that Are Potential MUIEs	Scenario Description, Characteristics	Are Any of the Converse Criteria Met? (Yes/No) Which?	PRA Results, Risk Significance, Other Notes	Screened Out?	
					Why?	Potential Negative Consequences of Screening Out IE?
Wind-Related Events	MU-IE-WIND-1	SBO and wind damage to SSCs	Yes: 1,2	All scenarios result in at least a single unit trip. Some scenarios are MU; others meet criterion 2. Wind events need to be further evaluated for MU potential. Wind events contribute 5% to the total single unit CDF.		

C.2.2 Results for the Identification of Sitewide IEs for the SFPs and DCS

The results of the sitewide IE identification for the SFP and DCS PRAs are presented in Section C.2.2.1 and Section C.2.2.2, respectively.

C.2.2.1 Results of the Identification of Sitewide IEs for SFPs

The scope and consequences considered by the SFP analysis (NRC, 2025a) were different than that for the reactors. In particular, the SFP analysis used a truncation time of 7 days for significant fuel uncovery (SFU) as the base case analysis and included a sensitivity case that considers events that were screened out by the 7-day sequence truncation. The sensitivity case explored the screened events and found that seismic events without leaks may significantly contribute to risk in the longer timeframe. Other events were determined not to contribute significantly to SFP risk, partially because of the redundancy of the SFP cooling systems between the two units. Note, the sensitivity case did not consider simultaneous damage to the reactors (and, unlike for the reactors, no sensitivity analysis was performed for the SFPs that considered the FLEX strategies).

For the base case, SFU was only reached before 7 days in cases when there was a leak, or a large amount of inventory lost from seismically induced sloshing. All loss of cooling events without sloshing did not reach SFU within 7 days and were screened out of the analysis. The specific events for the base case analysis that result in SFU within 7 days are:

- a seismic event leading to:
    - o no leak in either pool or the reactor (boiloff event) for the following operating cycle phases (OCPs) (due to the extent of sloshing):
        - ▪ SAR 1 and ASR1 – seismic bin 6
        - ▪ All OCPs – seismic bins 7 and 8
    - o small (82 gpm), medium (1,311 gpm), or large leaks in the reactor while connected to the SFP during an outage (OCPs SAR1 and ASR1) – seismic bins 3 through 8
    - o moderate (initial 200 gpm) or large (initial 1,500 gpm) leaks in the SFP liners
    - o combinations of the above leaks
- a non-seismic large reactor loss-of-inventory (LLOINV) with failure to inject, while the SFP is connected to the reactor (OCPs SAR1 and ASR1)

Table C-4 presents the results of the identification of potential sitewide IEs for the SFPs using the SFP Level 1 and Level 2 PRA (NRC, 2025a) and associated analyst knowledge and understanding. Results shown are for significant fuel uncovery frequency (SFUF), which is analogous to CDF for reactors. These results were taken from Table 3-34 in the SFP Level 1 and Level 2 PRA report (NRC, 2025a).

Table C-4 is similar to Table B-4 in Appendix B and is consistent with the associated guidance, which differs from that for the reactors. In particular, the analyst is directed to use the list of already-identified MUIEs as the basis for assessment and to record the following for each identified MUIE:

- whether the MUIE is also an IE that impacts the SFPs/DCS (yes or no)
- any refinement notes (such as different seismic event bins shown in italicized red font).
- risk relevance using Level 1 or Level 2 PRA results (with the analyst indicating their choice of metric)

The results for the SFP analysis base case (i.e., 7-day sequence truncation) are given in Table C-4, summarized as:

- Seismic events in bins 1–8 were identified as potential sitewide IEs.
    - However, seismic bins 1, 2, 3, and 8 make very small contributions to overall SFUF.
        - For bin 8, the seismic initiating event frequency is low.
        - For bins 1, 2, and 3, the probabilities of SFP failures (e.g., liner failures) are low because the SFP is robustly built. Also, the amount of sloshing for these bins was determined to be insignificant.
- All other events that were identified as potential MUIEs (e.g., LOOPs) for the reactors were found to be irrelevant to the SFPs.

Table C-4 Search for Potential Sitewide IEs that Impact SFPs Using SFUF Results

L3PRAs MUIEs	Relevant to SFPs? (Y/N)	Refinement/ Caveat Notes	Level 1/2 PRA Results, Risk Significance, Other Notes				Other Notes
			Risk Metric	% of Total	Risk Metric	% of Total	
LOOP (grid-related only)	No. As discussed above these events were screened out of the SFP analysis.		Unknown but a sensitivity study in the SFP analysis suggests that the contribution would be small.				
LOOP (switchyard-related only)							
LOOP (weather-related only)							
Seismic events	Yes	Bin 7	SFUF	40.5%			
	Yes	Bin 6	SFUF	37.6%			
	Yes	Bin 5	SFUF	15.5%			
	Yes	Bin 4	SFUF	5.1%			
	Yes	Bin 3	SFUF	0.9%			Small contribution to SFUF
	Yes	Bin 8	SFUF	0.4%			Small contribution to SFUF
	Yes	Bins 1 and 2	SFUF	0.0%			Negligible contribution to SFUF
Non-seismic LLOINV	Yes		SFUF	0.0%			Applicable during ASR/SAR (when shutdown unit is connected to the SFP).
Loss of NSCW	No						Screened out by the 7-day truncation time.

C.2.2.2 Results of the Identification of Sitewide IEs for DCS

Table C-5 presents the results of the identification of potential sitewide IEs for the DCS facility using the DCS PRA (NRC, 2024) and associated analyst knowledge and understanding. Results shown are for two representative risk metrics: (1) latent cancer fatality (LCF) risk to an individual within 10 miles of the site and (2) economic cost risk integrated across the region within 100 miles of the site. These results were taken from Table 3-10 in NRC (2024).

Table C-5 is similar to Table B-4 in Appendix B and is consistent with the associated guidance, which differs from that for the reactors. In particular, the analyst is directed to use the list of already-identified MUIEs as the basis for assessment and record the following for each identified MUIE:

- whether the MUIE is also an IE that impacts the SFPs/DCS (yes or no)
- any refinement notes (such as different seismic event bins shown in italicized red font).
- risk relevance using Level 1 or Level 2 PRA results (with the analyst indicating their choice of metric)

The results given in Table C-5 can be summarized as follows:

- Seismic events, especially those in Bins 5-7,[55] are identified as potential sitewide IEs that can:
    - impact dry cask loading in the auxiliary building
    - cause tipping and failing of casks on pads
- No other IEs are important to DCS because dry casks stored on the pad are purely passive systems and do not require electrical power or support systems.

Because the scope of the ISR task is limited to at-power events only, fuel handling, cask loading, and other similar events were screened out of the analysis.

Table C-5 shows that the only MUIEs relevant to the DCS facility are seismic events impacting either cask loading activities or cask storage on the pad, and these MUIEs make a very small contribution to DCS risk for the two representative risk metrics. In addition, the source term and overall risk from these events for DCS are far below those for the SFPs and reactors. As such, risk contributions from the DCS facility were excluded from the ISR task.

[55] Seismic bin 8 also impacts both reactors, the SFPs, and the DCS facility. However, the contribution to total single unit CDF for seismic bin 8 is low compared to other seismic bins.

Table C-5 Search for Potential Sitewide IEs that Impact the DCS Facility

L3PRA MUIEs	Relevant to DCS? (Y/N)	Refinement/ Caveat Notes	Level 1/2 PRA Results, Risk Significance, Other Notes				Other Notes
			Risk Metric	% of Total DCS Risk	Risk Metric	% of Total DCS Risk	
LOOP (grid-related only)	No						Screened out of the dry cask analysis.
LOOP (switchyard-related only)	No						
LOOP (weather-related only)	No						
Seismic events—failing auxiliary building during cask loading	Yes	Risk dominated by seismic bins 5–7	Individual LCF risk (0–10 miles)	0.01%	Economic cost (0–100 miles)	0.00%	Very small source term
Seismic events—casks on the pad tipping and failing	Yes	Risk dominated by seismic bins 5–7	Individual LCF risk (0–10 miles)	0.00%	Economic cost (0–100 miles)	0.00%	Very small source term

C.2.3 Summary of All Results for MUIEs and Sitewide IEs

Table C-6 and Table C-7 are provided below to provide perspective and understanding for the results of this sitewide dependency assessment task to identify potential sitewide IEs.

Table C-6 summarizes the CDF results from all modeled hazard categories in the single unit L3PRA. In particular, internal events (43 percent) and internal fires (41 percent) are the largest contributors to overall risk for the reactors, with seismic events (7.2 percent) and high winds (9.2 percent) also being significant contributors.

Table C-6 Summary of CDF Results from Level 1 PRAs for Single Reactor

Hazard	CDF (/rcy)	Percentage of Total CDF
Internal events	6.39E-5	42.4%
Internal floods	7.91E-7	0.5%
Internal fires	6.14E-5	41%
Seismic events	1.08E-5	7.2%
High winds	1.38E-5	9.2%
Total Single Unit CDF	**1.51E-4**	**100%**

Table C-7 combines the results from the previous tables given in this report. It shows all the IEs, for all hazards, that were identified as potential sitewide IEs for both the reactors and the SFPs and DCS. In summary, Table C-7 shows that:

- The following potential MUIEs are important to the reactors only:
  - LOOPs
  - fire events/scenarios given in Table C-7
- Seismic events are important to the reactors, SFPs, and DCS:
  - all bins are important to the reactors and the SFPs
    - bins 1-6 are the most important to the reactors
    - bins 5-7 are most important to the SFPs (with bin 7 having the largest contribution to risk)
  - bins 5-7 are important to DCS

These results were used as inputs to decisions made for later steps in the ISR task, such as which sitewide IEs to represent in the sitewide risk model. Other inputs (e.g., results of the Phase 2 sitewide dependency assessment) also were used in this decision-making process.

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, Internal Fires and External Hazards

Reactors				SFPs and DCS			
Potential MUIE	**Converse Criteria Met?**	**Refinement/Caveat Notes**	**CDF (/rcy) (%)***	**Relevant to SFPs/ DCS?**	**Risk Metric**	**% of Total^{+}**	**Refinement/Caveat Notes**
Internal Events							
Grid-Related Loss of Offsite Power (LOOP)	Yes (#1)	Sitewide LOOP would occur.	1.8E-5 (29%)	No	Unknown percentage; base case (7-day truncation) and sensitivity case (14-day truncation) for SFPs suggest that contribution would be small.		Screened out of base case SFP and DCS analyses.
Switchyard-Centered LOOP	Yes (#1)	Could result in sitewide or single unit LOOP.	1.0E-5 (16%)	No			
Weather-Related LOOP	Yes (#1)	Likely sitewide LOOP, but not definite.	9.0E-6 (14%)	No			
Loss of Nuclear Service Cooling Water (NSCW)	No	If cross-unit CCF is considered, dual-unit loss of NSCW can occur. The dominant loss of NSCW cutsets are from CCF pumps. This scenario is not recommended to be screened out.	8.8E-6 (14%)	No	Unknown percentage.		Screened out of base case SFP analysis (not applicable for DCS).
Interfacing System LOCA (ISLOCA) from Residual Heat Removal (RHR) Hot Leg Suction Lines	No	If cross-unit CCF of the RHR hot-leg suction isolation valves is considered, dual-unit ISLOCA can occur. This scenario is not recommended to be screened out—dominant CCF aspects and high-risk potential of dual-unit ISLOCA.	2.3E-7 (<1%)	No			

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and DCS			
Potential MUIE	**Converse Criteria Met?**	**Refinement/Caveat Notes**	**CDF (/rcy) (%)***	**Relevant to SFPs/ DCS?**	**Risk Metric**	**% of Total+**	**Refinement/Caveat Notes**
ISLOCA from RHR Cold Leg Injection Lines [Two IEs]	No	If cross-unit CCF of the RHR cold-leg injection isolation valves is considered, dual-unit ISLOCA can occur. This scenario is not recommended to be screened out - dominant CCF aspects and high-risk potential of dual unit ISLOCA.	8.4E-8 (<1%)	No			
Internal Floods							
1-FLI-TB_500_HI1	Yes; #3 possible	Flood in turbine building, main condenser	Not significant	No			
1-FLI-TB_500_LF	Yes; #3 possible	Flood in turbine building, CW expansion joint failure	2.1% of flooding CDF				
1-FLI-TB_500_LF-CDS	Yes; #3 possible	Flood in turbine building, piping failure	Not significant				
1-FLI-TB_500_HI2	Yes; #3 possible	Flood in turbine building, main condenser	Not significant				

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and DCS			
Potential MUIE	Converse Criteria Met?	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ DCS?	Risk Metric	% of Total+	Refinement/Caveat Notes
Internal Fires							
MU-IE-FRI-1	Yes; #1 and #2	Both MCRs evacuated (CCDP = 1); MCR abandonment scenarios	0.2%	No			MCR evacuation scenarios contributed less than 1% to CDF from internal fire events, and consequently, even less to the total plant CDF. However, with an MUCDF of 1.4E-07/rcy (their CCDP is 1.0), they should be retained in the MUCDF estimates. The remaining internal fire scenarios with MU potential were collected (mapped) into 3 generalized scenarios below. These three combined scenarios need to be evaluated further (including defining their representative scenarios) and should be addressed in MUCDF estimates.
MU-IE-FRI-2		Scenarios with shared areas between Units 1 and 2 (i.e., single unit fire scenarios beginning with "A" or "Y"), excluding MCR abandonment scenarios	16.3%				See caveat note above for MU-IE-FRI-1.

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and DCS			
Potential MUIE	Converse Criteria Met?	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ DCS?	Risk Metric	% of Total+	Refinement/Caveat Notes
MU-IE-FRI-3		Unit 1 fires that cascade to Unit 2	68.9%				See caveat note above for MU-IE-FRI-1.
MU-IE-FRI-4		Unit 2 fires that cascade to Unit 1	5.4%				See caveat note above for MU-IE-FRI-1.
Seismic Events							
MU-IE-EQK-1	Yes; #1 and #2	Seismic event in bin 1 (0.1–0.3g) occurs	12.0%	Yes (SFPs only)	SFUF**	0.0%	Negligible contribution to SFUF
MU-IE-EQK-2		Seismic event in bin 2 (0.3–0.5g) occurs	11.3%				
MU-IE-EQK-3		Seismic event in bin 3 (0.5–0.7g) occurs	15.0%	Yes (SFPs only)	SFUF**	0.9%	Small contribution to SFUF
MU-IE-EQK-4		Seismic event in bin 4 (0.7–0.9g) occurs	22.5%	Yes (SFPs only)	SFUF**	5.1%	
MU-IE-EQK-5		Seismic event in bin 5 (0.9–1.1g) occurs	20.8%	Yes (SFPs)	SFUF**	15.5%	
				Yes (DCS)	LCF risk 0–10 miles	See Notes	Two types of potential failure (both with very small source term): (a) failing auxiliary building during cask loading (bins 5-7: 0.01%), and (b) tipping and failing casks on the pad (bins 5-7: 0.00%).
MU-IE-EQK-6		Seismic event in bin 6 (1.1–1.5g) occurs	16.2%	Yes (SFPs)	SFUF**	37.6%	
				Yes (DCS)	LCF risk 0–10 miles	See Notes	See bin 5
MU-IE-EQK-7		Seismic event in bin 7 (1.5–2.5g) occurs	2.2%	Yes (SFPs)	SFUF**	40.5%	
				Yes (DCS)	LCF risk 0–10 miles	See Notes	See bin 5

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and DCS			
Potential MUIE	Converse Criteria Met?	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ DCS?	Risk Metric	% of Total+	Refinement/Caveat Notes
MU-IE-EQK-8		Seismic event in bin 8 (2.5g and above) occurs	0.02%	Yes (mostly SFPs)	SFUF**	0.4%	Small contribution to SFUF; even smaller contribution for dry cask storage risk
High Winds							
MU-IE-WIND-1	Yes; #1 and #2	SBO and wind damage to SSCs	100%	No			All wind scenarios modeled for Unit 1 are mapped into this scenario. A representative MU scenario can be assigned to this scenario. (If wind scenarios were considered individually, they could have been inadvertently screened out. Together, all wind scenarios contribute only 5% to the total plant CDF. They are mostly LOOPs, with insignificant damage to safety-related SSCs, even at high wind speeds.)

Table C-7 Summary of IE Screening for Internal Events, Internal Floods, and External Hazards (cont.)

Reactors				SFPs and DCS			
Potential MUIE	Converse Criteria Met?	Refinement/Caveat Notes	CDF (/rcy) (%)*	Relevant to SFPs/ DCS?	Risk Metric	% of Total+	Refinement/Caveat Notes
Low-power and shutdown (LPSD) conditions, SFP analysis							
Non-seismic LLOINV*	[redacted]	LPSD PRA for internal events only but out of scope for L3PRA project ISR task.	[redacted]	Yes (SFPs only)	SFUF**	0.0%	Applicable during OCPs ASR/SAR (when shutdown unit is connected to the SFP); out of scope for the L3PRA project ISR task.

* *Percentage of CDF for that specific hazard category.*

\+ *Percentage of SFUF from all hazards.*

*** SFUF: Significant fuel uncovery frequency (analogous to CDF)*

C.3 Selection of MUIEs and Sitewide IEs to Represent

Several factors were considered for the ISR task in selecting which MUIEs and/or sitewide IEs to use in MU risk calculations. The information presented in earlier sections of this appendix, such as percentage of independent radiological source contributions to risk and number of radiological sources affected by the initiator, is an important factor to this selection process. Resource constraints for the overall L3PRA project was another important factor.

In selecting MUIEs and sitewide IEs, the focus was on the initiating events identified in Phase 1 of the sitewide dependency assessment that can impact two reactors (as well as those that can impact either the SFPs or DCS). Table C-8 below shows the IEs that have MU and/or sitewide impact.

The IEs shown in Table C-8 were addressed in the MUCDF estimations. Note that, from the Phase 1 sitewide dependency assessment, the only relevant initiators for the SFPs are seismic events. Consequently, when sitewide scenarios are developed and sitewide risk estimated, the SFPs only contribute to results associated with seismic events.

Table C-8 List of IEs that Have Potential Multi-Unit or Sitewide Impacts

No.	Scenario Name	Scenario Description	MU Scenario Characteristics
1	MU-IE-LOOPGR	Grid-related LOOP	SBO and AC power recovery failure
2	MU-IE-LOOPPC	Plant-centered LOOP	SBO and AC power recovery failure
3	MU-IE-LOOPSC	Switchyard-centered LOOP	SBO and AC power recovery failure
4	MU-IE-LOOPWR	Weather-related LOOP	SBO and AC power recovery failure
5	MU-LONSCW	Loss of NSCW	Loss of NSCW in both units
6	MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned with CCDP =1
7	MU-IE-FRI-2	U1 and U2 shared (A+Y) area fires	at least MU LOOP
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least MU LOOP
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least MU LOOP
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic bin 1
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic bin 2
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic bin 3
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic bin 4
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic bin 5
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic bin 6
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2-unit SBO and major structural damage (seismic bin 7) with CCDP =1
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and Major structural damage (seismic bin 8) with CCDP = 1
18	MU-IE-WIND-1	SBO and SSC wind damage	SBO and WIND damage to SSCs

C.4 Calculation of MUIE and Sitewide IE Frequencies

There are differences between single unit IEs in how they were calculated and the data used in those calculations. Some of the MUIEs or sitewide IEs not only impact the entire reference site but also were initially developed as sitewide frequencies. All IE frequencies for external hazards (e.g., seismic events) were developed in this way. However, per PRA convention, even these IE frequencies were reported in “per-reactor-critical-year” units.[56] Consequently, the originally determined frequency for these IEs was used directly in MU risk calculations. In addition, the original IE frequency was used for certain fire scenarios (e.g., MCR abandonment scenarios and fires that cascade from one unit to another).

Other MUIEs or sitewide IEs were adjusted for MU risk calculations. These IEs are:

- LOOP
- loss of nuclear component service water

The calculation of the frequencies for each of these IE types is discussed below.

C.4.1 LOOPs

As described above (and as typical for PRAs), there are four types of LOOPs to be addressed:

- grid-related LOOP (LOOPGR)
- plant-centered LOOP (LOOPPC)
- switchyard-centered LOOP (LOOPSC)
- weather-related LOOP (LOOPWR)

A variety of approaches have been used or proposed for developing MUIE frequencies (MUIEFs) for LOOPs. Examples of such approaches are given in IAEA (2019, 2021a).

Section 2 of the L3PRA project’s report on the reactor, at-power, Level 1 PRA for internal events (NRC, 2022d) outlines the approach used to develop the single unit IE frequencies. As is shown in Table 2-1 of NRC (2022d), the basis for the LOOP frequencies is the 2010 update to NUREG/CR-6928 (INL, 2007).

Various approaches have been used or proposed for developing MUIEFs, all of which require IE data to be separated into the LOOP categories above. For example, Sections 5.2.5.2 and 5.2.5.3 of IAEA (2019) discuss three different approaches, all involving re-analysis of LOOP data. Section 2.3.1.4 of EPRI (2021a) documents the results of analysis of international LOOP data, producing generic “fractional adjustments for MU initiators,” or conditional probabilities of an MUIE given the occurrence of a single unit IE. Table 2-3 in EPRI (2021a) shows the following fractions for switchyard-centered, weather-related, and grid-related LOOPs[57]:

[56] Typically, a capacity factor is used with IE frequencies that have been developed in this way. The L3PRA project did not use capacity factors in its PRAs. However, since the capacity factor for the reference plant is high (i.e., 0.93), the difference between reactor-critical-year and reactor-calendar-year is well within uncertainty bounds.

[57] Note that the EPRI report (EPRI, 2021a) uses different acronyms than the L3PRA project for the different categories of LOOP.

- LOOPSC: 0.5
- LOOPWR: 1.0
- LOOPGR: 1.0

The approach used for the L3PRA project's ISR task is similar to that used in EPRI (2021) in that MU conditional probabilities were used to convert single unit IE frequencies into MUIEFs. However, the MU conditional probabilities used in the ISR task are taken from Table 17 of the 2021 update (INL, 2021) of Idaho National Laboratory's (INL's) "Analysis of Loss-of-Offsite-Power Events Update" report (INL, 2007). INL (2021) used only U.S. data (2006 through 2020) to develop MU conditional probabilities (unlike the EPRI report's use of international data). The ISR task uses the mean values shown in INL (2021), as replicated in Table C-9 below along with the resulting MUIE frequencies (MUIEFs).[58] Note that the INL data analysis, unlike the EPRI report's analysis, indicates that even LOOPPCs can result in an MU event.

The MUIE frequency is calculated through use of a multiplier. A Unit 2 multiplier is introduced to calculate a two-unit scenario initiating event frequency. This multiplier was multiplied by the Unit 1 IE frequency (U1IEF) to obtain an MUIEF. The multiplier is 1.0 if the Unit 1 IE also causes a Unit 2 trip. If a fraction of the Unit 1 initiating events causes a Unit 2 trip, the multiplier is equal to the fraction. The multiplier cannot be greater than 1.0.

Table C-9 MU Conditional Probabilities and Resulting MU Multipliers for MUIEs

	Scenario Name	U1IEF	MU Multiplier	MUIEF
1	MU-IE-LOOPGR	1.23E-02	0.500	6.15E-03
2	MU-IE-LOOPPC	1.93E-03	0.056	1.07E-04
3	MU-IE-LOOPSC	1.04E-02	0.269	2.80E-03
4	MU-IE-LOOPWR	3.91E-03	0.625	2.44E-03

C.4.2 Loss of Nuclear Component Service Water

The IE frequency for the loss of NSCW that was used in the L3PRA project is based on a CCF analysis. As such, the frequency of a multi-unit loss of NSCW also was developed via CCF analysis.

For the MUCDF results developed at this time, complete dependency was assumed between the NSCW pumps such that the single unit IE frequency is used as the MUIEF, too (i.e., a multiplier of 1.0). Appendix H provides further discussion on the development of the MUIEF for loss of NSCW.

C.4.3 Overall Results for MUIE and Sitewide IE Frequencies

The final MUIE or sitewide IE frequencies used in the ISR task are shown in Table C-10.

[58] Note that the ISR task uses MU conditional probabilities based on data in 2021 updated report. However, the L3PRA project's PRA models have a freeze date of 2012 so they use an earlier version of LOOP data for the single unit IE frequencies.

Table C-10 MU and Sitewide Initiating Event Frequencies

	Scenario Name	U1IEF (/rcy)	MU Multiplier	MUIEF (/rcy)
1	MU-IE-LOOPGR	1.23E-02	0.500	6.15E-03
2	MU-IE-LOOPPC	1.93E-03	0.056	1.07E-04
3	MU-IE-LOOPSC	1.04E-02	0.269	2.80E-03
4	MU-IE-LOOPWR	3.91E-03	0.625	2.44E-03
5	MU-LONSCW	3.47E-05	1	3.47E-05
6	MU-IE-FRI-1	1.50E-07	1	1.50E-07
7	MU-IE-FRI-2	3.40E-02	1	3.40E-02
8	MU-IE-FRI-3	9.10E-03	1	9.10E-03
9	MU-IE-FRI-4	9.10E-03	1	9.10E-03
10	MU-IE-EQK-1	1.60E-03	1	1.60E-03
11	MU-IE-EQK-2	2.20E-04	1	2.20E-04
12	MU-IE-EQK-3	4.80E-05	1	4.80E-05
13	MU-IE-EQK-4	1.30E-05	1	1.30E-05
14	MU-IE-EQK-5	4.30E-06	1	4.30E-06
15	MU-IE-EQK-6	1.90E-06	1	1.90E-06
16	MU-IE-EQK-7	2.50E-07	1	2.50E-07
17	MU-IE-EQK-8	2.30E-09	1	2.30E-09
18	MU-IE-WIND-1	8.89E-03	1	8.89E-03
				7.45E-02

*rcy – reactor-critical-year

APPENDIX D
IDENTIFICATION OF SHARED PHYSICAL RESOURCES AND SHARED OR CONNECTED AND SYSTEMS, STRUCTURES, AND COMPONENTS

This appendix presents the results for the Phase 2 sitewide dependency assessment as part of the integrated site risk (ISR) task.

D.1 Approach for Phase 2 Sitewide Dependency Assessment

Section B.6 describes the approach used for the Phase 2 sitewide dependency assessment. Based on this approach, there are three types of results for this sitewide dependency assessment:

1. shared physical resources (Section D.2)
2. shared or connected SSCs (Section D.3)
3. assessment of coupling between the two reactor units (Section D.4)

The results given below address the two reactors, the spent fuel pools (SFPs), and dry cask storage (DCS). The "base case" results correspond to the overall freeze date for the L3PRA project of August 2012 (with a few exceptions). In addition, sensitivity analyses for FLEX strategies have been performed for the two reactors, as documented in the single unit L3PRA project PRA reports. A similar sensitivity analysis for FLEX strategies was not performed for the SFPs.

The assessments were performed in succession, each analyst building on the previous assessment. The order of inputted results from analysts was:

1. internal events Level 1 PRA for the two reactors
2. internal floods Level 1 PRA for the two reactors
3. fire, seismic, and wind Level 1 PRAs for the two reactors
4. FLEX strategies[59] for the two reactors
5. Level 2 PRA for the two reactors
6. all hazards and Level 1 and 2 PRAs for the spent fuel pools (SFPs) and the DCS facility

Worksheets such as those shown in Table B-9 and Table B-10 were used to document the identification of identical components between the two reactors, then between the SFPs/DCS and the two reactors, respectively. The tables and associated approach used for this identification is similar to that used for shared physical resources (Section B.6.1) and shared or connected SSCs (Section B.6.2). Illustrative examples are shown in red font in Table B-9 and Table B-10.

It was expected that the majority of CCFs relevant to MU risk would be identified from the single unit, Level 1 internal events PRA model. However, it was recognized that additional CCF groups could be modeled in Level 1 PRAs for other hazards or in the Level 2 PRAs. Consequently, analysts were asked to document any risk significant CCF groups for other hazards (e.g., fire or

[59] 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.

seismic) even if these groups had already been identified from the Level 1 internal events PRA. Such information could be helpful in later screening for the ISR task.

Identical components that are modeled in both reactor units were considered for modeling cross-unit CCFs. Two different types of CCFs were identified and documented for potential consideration in the ISR task:

- CCFs that are already modeled in the L3PRA project PRA models

- new CCFs involving identical components across the two reactors units (or between the reactors and the SFPs or dry cask storage)

Note that this identification did not include CCFs for components not already included within the single reactor PRA models (which is a scope choice made for the L3PRA project).

For CCFs that were already modeled in the single unit base PRA, there were two cases: (1) the existing CCF group size was also appropriate for a multi-unit risk model, and (2) the existing CCF group size had to expanded for the multi-unit risk model. For example, if there were CCFs already modeled for a system that is shared by both reactor units (e.g., CCFs of service water pumps) and the success criteria is not changed when going from the single unit to the multi-unit model, then no expansion of the common cause component group (CCCG) is needed. However, if the CCFs in the single unit PRA model are not in a shared system (e.g., CCF of emergency diesel generators [EDGs]), the CCCG would need to be expanded.

D.2 Results for the Identification of Shared Physical Resources

The results for the identification of potential dependencies for the category of “shared physical resources” are given below. Physical resources shared by the two reactors are provided first, followed by physical resources shared between the SFPs/DCS and the two reactors.

D.2.1 Shared Physical Resources Between the Two Reactors for Level 1 and 2 PRAs

Table D-1 summarizes the combined results for the two reactors for all hazards, both Level 1 and 2 PRAs. This table contains the following columns:

- identified dependencies
- relevant IEs [initiating events] and MUIEs [multi-unit initiating events]
- relevant hazards
- notes
- keys inputs to modeling decisions

Based on the information available to the project team, potential dependencies between the two reactors regarding shared physical resources include water sources (e.g., fire water storage tanks [FWSTs] outside the auxiliary building, and the demineralized water storage tank [DWST]) that can be used with the B.5.b pumps when implementing extensive damage mitigation guidelines (EDMGs) for Level 2 PRA response.

Based on information provided in the reference plant Final Integrated Plan (FIP), it is assumed that there is sufficient diesel fuel and refueling trucks to support FLEX strategies for all relevant radiological resources (i.e., both reactors and both SFPs).[60]

In summary, Table D-1 shows that there are only three physical resources shared between the two reactors. Two are related to electric power needs: (1) 230 kV and 500 kV switchyards and (2) the alternate switchyard. The main switchyards (and offsite power sources) were identified in the Phase 1 identification of sitewide IEs. Consequently, this dependency was addressed in the multi-unit risk model as a sitewide IE. The alternate switchyard, on the other hand, can be used to supply power to only one of the two units (and is currently credited in the single unit PRA model). So, if relevant, addressing this dependency would require modeling an asymmetry between the two reactor units (i.e., only one unit can credit use of the alternate switchyard).

The third shared resource is water; namely, water tanks that are used with B.5.b pumps in modeling EMDG strategies in response to Level 2 PRA scenarios. At present, the needed volume of water for success of such EDMG strategies is assumed to be equivalent to both FWSTs. However, the smaller volume DWST is indicated to be an option, too. It is not currently known whether EDMG strategies can be successful with the smaller volume DWST. Also, it is not known if other water sources are available (and what procedures, training, etc., would support their use).[61]

[60] For example, the reference plant FIP indicates that there are three diesel refueling trucks.

[61] The reference plant Technical Support Guideline has a table for “Water Sources” but the DWST is not included in the table.

Table D-1 Shared Physical Resources Between the Two Reactor Units

Identified Dependencies	Relevant IEs and MUIEs	Relevant Hazards	Notes	Key Inputs for Modeling Decisions
230kV and 500kV Switchyards	All LOOPs	Internal events, seismic events, high winds, etc.	Documentation indicates that there are 230 kV and 500 kV switchyards. However, based on available information, there does not appear to be separation between them (i.e., there is one big switchyard for both units). Under normal operation, Unit 1 Division I and Unit 2 Division II are fed by one offsite source, while Unit 1 Division II and Unit 2 Division I are fed from the other offsite source. The two offsite power sources are separated physically as they leave the 230 kV substation and are arranged so that no single event, such as a falling line, tower, or other structure, will damage both lines. The following statement is in the electrical system notebook, "*Since no major equipment, electrical buses, or EDGs are shared between Units 1 and 2, the impact on either of a loss of offsite power occurring simultaneously at both units can be analyzed by two independent Unit 1 and 2 models*."	These dependencies also were captured in Phase 1, identification of sitewide IEs. These dependencies will be addressed via sitewide IEs.
Alternate switchyard	Plant-centered, switchyard, and consequential LOOPs	Internal events	Can only supply one unit at a time. The alternate switchyard is already assumed to be unavailable for weather- and grid-related LOOPs. May have limited effect since plant and switchyard LOOPs are less likely to be MUIEs.	This is potentially important dependency can only be captured in development of the multi-unit PRA model. Likely, Unit 1 will be credited with use of the alternate switchyard, and Unit 2 will not. This results in an asymmetry between the two reactor units.

Table D-1 Shared Physical Resources Between the Two Reactor Units (cont.)

Identified Dependencies	Relevant IEs and MUIEs	Relevant Hazards	Notes	Key Inputs for Modeling Decisions
North and South Fire Water Storage Tanks (FWSTs)	Level 2 scenarios	Internal events and internal floods, seismic events, etc.	The Level 2 PRA report (NRC, 2022b) describes the equipment and resources needed to implement Extensive Damage Mitigation Guidelines (EDMGs) in response to post-core-damage scenarios.	The volume from **both** FWSTs (total of 600,000 gallons) is used to implement the associated EDMG strategies. The demineralized water storage tank (DWST) can be used as a water source; however, the DWST has a smaller volume.

Notes:

a. Note that the ultimate heat sink is not shared between the two units because their nuclear service cooling water (NSCW) systems are completely separated with no shared intake structure. However, the well water storage tank and well pumps are shared between the cooling tower basins of both units. The combined capacity of two cooling tower basins at each unit is sufficient to last 27 days under worst case heat load conditions. Note that the NSCW systems do share common component types and procedures.

b. The model was searched for Unit 2 basic events, which revealed only Unit 2 EDG basic events and Unit 2 instrument air system isolation valve 2-2401-510.

c. A focused search was performed of the plant's system notebooks to identify shared physical resources, system-crossties, etc. Note that some of the system notebooks do not acknowledge Unit 2 at all (main feedwater/condensate, turbine plant closed cooling water, circulating water). The following shared resources were found:

   i. Each unit has one hydrogen recombiner; however, there is also a common recombiner that is served by either unit's auxiliary component cooling water system.

   ii. It appears that the same fans powered from Unit 2 motor control centers provide room cooling for both units component cooling water pumps (potential documentation error). Note that room cooling requirements were screened out and, therefore, are not included in the PRA.

   iii. The documentation does not describe if the units share the same circulating water bay or if they are separated.

D.2.2 Shared Physical Resources Between the SFPs and DCS with the Two Reactors for Level 1 and 2 PRAs

The assessment of potential dependencies between the SFPs and DCS with the reactors is tied to the risk consequences used for the L3PRA project. The discussion below summarizes the results of the assessments for the SFPs and DCS with respect to shared physical resources. The discussion given in this section is also relevant to Section D.3.2 for the assessment of shared and connected SSCs between the SFPs and the two reactors.

It should be noted that FLEX strategies have not been addressed for the SFPs. In particular, a sensitivity case similar to that of the reactors was not developed for the SFPs in the L3PRA project.

There are no shared physical resources between DCS and the two reactors. The DCS facility is a separate facility that does not require any external resources (e.g., electric power or cooling water) to prevent fuel damage. The passive design of the casks and the facility are sufficient to maintain necessary cooling and fuel configuration, even in the case of the most damaging seismic event considered in the L3PRA project.[62]

Table D-2 shows the physical resources shared between the SFPs and the reactors. Table D-2 contains the following columns:

- identified dependencies
- reactor MUIEs
- relevant hazards
- modeling or screening notes

As part of the L3PRA project, two analyses were performed for the SFPs: (1) the base case, for scenarios that lead to SFP uncovery within 7 days, and (2) a sensitivity case that relaxes the 7-day truncation time. For the base case, the only events that lead to uncovery within 7 days are those that result in inventory loss through a leak or sloshing out of the SFPs (i.e., mostly seismic events and a non-seismic reactor-side loss of inventory [LOINV] with the gates open). Further assumptions or scope limitations are:

- It is assumed that all seismic events result in a loss of offsite power.

- For seismic events, the normal cooling system for the SFPs, the spent fuel pool cooling and purification system (SFPCPS), is assumed to be lost and, therefore, is not credited since it has no emergency function during an accident and the suction line uncovers after the loss of approximately 4 feet of water (which happens immediately from sloshing for higher seismic bins, and soon after for other bins when a leak is present).

- Only strategies given in the EDMGs are credited. As stated in Section 7.5.2, the L3PRA project SFP PRA models two types of EDMG strategies: (1) an “internal strategy” (i.e., equipment predominantly located in the vicinity of the refuel floor in combination with

[62] The exception to these statements is for the very short amount of time during cask loading where SFP water is circulated through the cask. However, there are several backup strategies for restoration of cooling, including returning the cask to the SFP.

installed systems) and (2) an “external strategy” (i.e., use of on-site portable equipment and installed tanks that are deliberately remote from the refuel floor).

- For the base case, passive (e.g., gravity-feed) strategies are not credited because the flowrates for these strategies are too low to mitigate the loss of SFP inventory events that can lead to fuel uncovery within 7 days.

- According to the reference plant’s EDMGs, the internal EDMG strategy cannot be used if the SFPs are inaccessible or if there is excessive loss of SFP inventory (e.g., greater than 500 gpm leakage).

- For the non-seismic LOINV event, the following is assumed:

    - Offsite power is available.
    - SFPCPS cooling is available.

Both the base analysis and the sensitivity analysis credit the same two strategies from the EDMGs (i.e., the internal and external strategies mentioned above). The relevant EDMG strategies are detailed in the reference plant procedure, “Emergency Management Guideline (EMG).” The procedure describes multiple options for restoring level for the SFPs (e.g., multiple locations for standpipe valves) and using two different approaches (i.e., makeup or spray). Because the base case and the sensitivity case use different assumptions, different parts of the described strategies (and different associated equipment) are used in the base and sensitivity cases.

Table D-2 Physical Resources Shared Between the SFPs and the Reactors

Identified Dependencies	Rx MUIEs	Relevant Hazards	Modeling or Screening Notes
Level 1 and 2 PRAs – Base and Sensitivity Analyses			
230kV and 500kV Switchyards	All LOOPs	Internal events, seismic events	*Internal EDMG strategy:* Specifically, electric power is needed to operate the NSCW systems in order to replenish SFP inventory. *Sensitivity case only:* Offsite power is used to facilitate normal cooling of SFPs via NSCW standpipes.
Ultimate heat sink and associated intake structure	All LOOPs	Internal events, seismic events	*Internal EDMG strategy:* Specifically, the water inventory in the NSCW systems is needed to replenish SFP inventory via NSCW standpipes.
Water storage tanks: FWSTs (2) and DWST (1)	All LOOPs	Internal events, seismic events	*External EDMG strategy:* Specifically, the water inventory in the FWSTs or DWST is needed to replenish SFP inventory using a B.5.b pump.
Water supply for refilling FWSTs and DWST	All LOOPs	Internal events, seismic events	*External EDMG strategy:* Specifically, the water inventory in the FWSTs or DWST may need to be replenished.
Various water tanks inside the plant (e.g., RWSTs, RMWSTs, or DWST)	All LOOPs	Internal events, seismic events	*Sensitivity case only: These tanks are used for the gravity-feed strategy.*
FLEX Strategies			
FLEX pumps and associated equipment	All LOOPs	Internal events, seismic events	
Various water sources	All LOOPs	Internal events, seismic events	

DWST: demineralized water storage tank
FWST: fire water storage tank
RWST: refueling water storage tank
RMWST: reactor makeup water storage tank

D.2.2.1 Results for Base Case: SFP Level 1 and 2 PRAs

The internal EDMG strategy uses two firewater standpipes which are respectively fed by the two NSCW systems from the NSCW basins (which have a huge inventory). Two standpipes, one from each of the two NSCW systems, are needed for this strategy. Consequently, the SFPs share:

- the fire protection system (of which the standpipes are part)

- the ultimate heat sink with the two reactors, including the NSCW pumps, NSCW tower fans, and other associated components and structures

- electric power sources (either offsite power or EDGs), which are needed to operate the NSCW systems

Additional dependencies between the SFPs and the two reactors that are related to this strategy are identified in Section D.3 for shared or connected SSCs. Also, the reference plant EDMG procedure states that this strategy requires an operator manual action to open valves. These valves are locked, requiring a key or bolt cutters. The assessment of Phase 3 potential sitewide dependencies considers this operator action within the category of human and organizational resources (see Appendix F).

The external EDMG strategy uses either the FWSTs or the DWST (both of which are located outside plant buildings) with a B.5.b pump (which is addressed in Section D.3.2) and associated hoses. There are two FWSTs (North and South) and a single DWST. Since the Level 2 PRA for the single unit reactor credits EDMG strategies that use the FWSTs, Table D-4 shows these tanks as dependencies with the two reactors. In addition, these tanks may need to be refilled.

In summary, for the base case SFP PRA, there is sharing of the following physical resources between the SFPs and the reactors:

- electric power sources (i.e., switchyards)
- ultimate heat sink, NSCW basins, and NSCW intake structures[63]
- water tanks outside the security fence
    - FWSTs
    - DWST

- water supplies for refilling water tanks

Other potential dependencies mentioned above are addressed in the next section for shared or connected SSCs or, in some cases, in the Phase 3 assessment of potential sitewide dependencies.

D.2.2.2 Results for Sensitivity Case: SFP Level 1 and 2 PRAs

The sensitivity case for the SFPs addresses scenarios that extend beyond the 7-day truncation time. In this sensitivity case, the SFP analysis models additional SSCs beyond those included in

[63] The intake structure is addressed in Section 12D.3.2 for shared and connected SSCs.

the base case, including the use of offsite power or the EDGs to facilitate normal cooling of the SFPs with the SFPCPS or, if that strategy fails, use of a gravity makeup strategy that involves one of several water tanks. The gravity makeup strategy includes the assumption that valves can be operated manually if the normal motive force (via either instrument air or electric power[64]) is lost, in order to restore water level and normal cooling. (The assessment of Phase 3 potential sitewide dependencies considers this operator action within the category of human and organizational resources.)

Table D-2 shows the potential dependencies between the SFPs and the two reactors for both the base and sensitivity cases. For the sensitivity case of the SFP PRA, there is sharing of the following physical resources between the SFPs and the reactors:

- switchyards (for offsite power)
- ultimate heat sink, NSCW basins, and NSCW intake structures
- water tanks inside the plant, such as:
    - refueling water storage tanks (RWSTs)
    - reactor makeup water storage tanks (RMWSTs)
    - DWST

Other potential dependencies mentioned above (e.g., the EDGs) are addressed in the next section for shared or connected SSCs or, in some cases, in the Phase 3 assessment of potential sitewide dependencies.

D.3 Results for the Identification of Shared or Connected SSCs

The results for the identification of potential dependencies for the category of “shared or connected SSCs” are given below.

D.3.1 Shared or Connected SSCs Between the Two Reactors

Table D-3 summarizes the combined results for the two reactors for all hazards, both Level 1 and 2 PRAs, and FLEX strategies. This table contains the following columns:

- category (of potential sitewide dependency)
- identified dependencies
- relevant hazards and MUIEs
- notes
- keys inputs to modeling decisions

The following are assumptions for potential dependencies between the two reactors regarding shared or connected SSCs:

- As noted in Table D-3, cross-unit internal floods for the control buildings were screened out per walkdowns and reviews of building layouts.

[64] With currently available plant information, it has not been possible to determine how these valves are powered.

- As noted in Table D-3, the utility's fire PRA did not consider smoke from fires in the cable spreading rooms to be sufficient to cause main control room (MCR) abandonment. The L3PRA project modeling also uses this understanding.

In summary, Table D-3 shows that:

- The only common systems or components between the two reactors are the B.5.b pumps and associated equipment needed for Level 2 PRA scenarios.

- The only common or shared structure between the two reactor units is the FLEX building.[65] However, since FLEX building has been specifically designed and constructed to withstand external events, its failure is not considered for MU risk for either external or internal events.

- There are several buildings that are connected between the two units: (a) auxiliary buildings, (b) control buildings (including the technical support center [TSC]), (c) MCRs, (d) cable spreading rooms, and (e) turbine buildings.

    - None of these building connections are considered important dependencies for internal events and internal floods PRAs.

    - All these building connections are flagged as being potentially important for seismic events but are considered to be best addressed in Phase 3 of the sitewide dependency assessment.

    - The connection between the MCRs of Units 1 and 2 is an important dependency for certain fires that could produce enough smoke to prompt abandoning both MCRs.

    - Connections between the auxiliary buildings, control buildings, and turbine buildings are identified as being potential important dependencies for fire events. There are multiple scenarios in the single unit, base fire PRA for which a fire in Unit 2 propagates and leads to core damage in Unit 1. The specific fire locations and associated equipment and connections for these scenarios are not well-understood at this time due to limited available documentation of the fire PRA.

[65] Although the reactors share the fuel handling building, it is not noted here since the SFPs are considered a separate radiological source in this dependency assessment.

Table D-3 Shared or Connected SSCs Between the Two Reactors

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions
Level 1 and 2 PRAs				
Shared or connected systems and components	EDGs[a]	All LOOPs	The plant can crosstie an EDG to the opposite unit. However, the L3PRA does not credit this because it is not proceduralized.[b]	For the reasons described in the previous column and the table notes, this potential dependency is not included in the multi-unit risk model.
	B.5.b pumps and associated equipment (e.g., trailers, hoses, vehicle(s) to pull the trailers)	Level 2 PRA scenarios	There are two B.5.b pumps (and associated equipment) to implement EDMG strategies. However, one B.5.b pump is stored nearby (in the warehouse), while the other is at the fire training facility (farther away).	While, in principle, two B.5.b pumps for two reactors should be sufficient, it is not known if there is adequate time and other resources to use the second B.5.b pump that is located farther away from the reactors and associated connection points.
	No other shared or connected systems and components were identified.[c]			
Shared or connected structures	Auxiliary buildings	Internal events, internal fires, internal floods, seismic events	Unit 1 and Unit 2 auxiliary buildings are connected but separated by walls and doors. Safety-related equipment is further separated by placement away from the opposite unit. See Appendix C, Section C.3.1, of the internal flooding PRA report (NRC, 2022e) for more information on building layout.	Because of the separation noted in the previous column, the connections between the auxiliary buildings for Units 1 and 2 are not expected to be an important dependency for internal events and internal floods. The connections between the auxiliary buildings for Units 1 and 2 may be relevant for MU fire PRA.[d]

Table D-3 Shared or Connected SSCs Between the Two Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions
				The connections between the auxiliary buildings for Units 1 and 2 are likely to be relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	Control building	Internal events, internal fires, internal floods, seismic events	Unit 1 and Unit 2 share the control building although there is some separation by walls and doors. There is one shared room on the upper level (Level 3) with normal building air conditioning equipment (not the main control room [MCR] heating, ventilation, and air-conditioning [HVAC]). Possible flood propagation paths exist between units, but cross-unit internal flood scenarios were screened out of the single unit internal flood PRA.	Because of the separation noted in the previous column, the sharing of the control building for Unit 1 and Unit 2 is not expected to be an important dependency for internal events. For internal floods, the previous column states that cross-unit internal floods for the control building were screened out of the single unit PRA. Per the scope decisions for the L3PRA project, they also will not be addressed as part of the ISR task. The control building connections may be relevant for MU fire PRA.[d] The control building connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).

Table D-3 Shared or Connected SSCs Between the Two Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions
	MCRs	Internal events, internal fires, internal floods, seismic events	The Unit 1 and Unit 2 MCRs share the same space on Level 1 of the control building. The Unit 1 and Unit 2 control rooms are separated by a partial wall (partition). Unit 1 and 2 control rooms have separate HVAC equipment. Internal flooding is screened due to lack of flood sources and low likelihood of other flood sources propagating to the control rooms. A fire (that results in smoke to reach abandonment criteria) in either MCR would result in dual-unit abandonment.	Because of the separation stated in the previous column, the connections between the MCRs for Unit 1 and Unit 2 are not expected to be an important dependency for internal events and internal floods. Dual-unit MCR abandonment scenarios that involve a fire in either of the two MCRs will be considered in the MU risk model. Also, the connections between the MCRs are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	Technical support center	Internal events, internal fires, internal floods, seismic events	The technical support center (TSC) is common to both units and is located in the control building that is shared by Unit 1 and Unit 2. Since the TSC is located in the shared control building, it is evaluated the same way as described above for the connected control buildings.	See the above evaluation for the connected control buildings.
	Cable spreading rooms	Internal fires	There are two cable spreading rooms for each unit (four rooms in all). The MCRs are on Level 1. There is a cable spreading room for each unit at the elevation above, Level 2, and one at	Because of the separation noted in the previous column, the connections between the cable spreading rooms for Unit 1 and Unit 2 are not expected to be an

Table D-3 Shared or Connected SSCs Between the Two Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions
			the elevation below, Level A. The rooms are separated between units with a door between them. A drawing of Level A was used for this assessment and Level 2 is assumed to be similar (but the project does not have that drawing). Floor and ceiling penetrations are sealed with foam. In addition, the utility fire PRA considered potential MCR abandonment scenarios involving sources of smoke outside the MCR. However, the information supporting the fire PRA states that there are no fires, outside of MCR fires, that produce sufficient smoke that is transported to the MCR and could cause a habitability concern.	important dependency for internal events and internal floods. Per information supporting the utility's fire PRA, smoke from fires in the cable spreading rooms will not cause MCR abandonment. However, like the connected auxiliary and control buildings, the connections between the cable spreading rooms may be relevant for MU fire PRA.[d]
	Turbine buildings	Internal events, internal fires, internal floods, seismic events	Unit 1 and Unit 2 turbine buildings are connected, but most areas are separated by walls and doors. The turbine deck area is open between units, but the Unit 1 and Unit 2 equipment are physically separated.	Because of the separation noted in the previous column, the connections between the turbine buildings for Unit 1 and Unit 2 are not expected to be an important dependency for internal events and internal floods. However, like the connected auxiliary and control buildings and the cable spreading rooms, the connections between the turbine buildings may be relevant for MU fire PRA.[d]

Table D-3 Shared or Connected SSCs Between the Two Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions
				The connections between the turbine buildings are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	Fuel handling building	Internal events, internal fires, internal floods, seismic events	The fuel handling building is common to Units 1 and 2. The fuel handling building houses both units' spent fuel pools (SFPs), which are normally connected through the cask loading pit.	Because the SFPs are considered a separate radiological source under the ISR task, the fuel handling building is not considered a shared structure for this analysis. The shared fuel handling building is likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations). The two concerns are habitability on the SFP floor (with respect to operator actions) and structural damage.
	FLEX storage building	Internal events, internal fires, internal floods, seismic events, wind events	Single building storing the portable FLEX equipment for both units (per the reference plant FIP).	The FLEX building is designed and constructed to withstand external hazards. Consequently, it is unlikely that it would fail for any of the modeled internal and external hazards, with the possible exception of some large seismic events (i.e., high seismic bins).

Notes:

Table D-3 Shared or Connected SSCs Between the Two Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Key Inputs for Modeling Decisions

a. The model was searched for Unit 2 basic events, which revealed only the Unit 2 EDG basic events and Unit 2 instrument air system isolation valve 2-2401-510.
b. In addition, an HRA-focused plant site visit confirmed that the strategy for cross-tying EDGs would need to be developed by engineers in the TSC using electrical drawings. Also, interviews of operations managers revealed that, if a mistake is made and the EDG being cross-tied is the only available source of power, there is a chance that both units can lose power. It is for that reason that this option was pulled out of procedures and MCR operators' responsibilities. One operations manager who was interviewed said that the only context in which he would authorize this cross-tie option is if one unit had offsite power and did not need the EDG.
c. A focused search was performed of the plant's system notebooks to identify shared physical resources, system crossties, etc. Note that some of the system notebooks do not acknowledge Unit 2 at all (main feedwater/condensate, turbine plant closed cooling water, circulating water). The following shared resources were found:
   I. Each unit has one hydrogen recombiner; however, there is also a common recombiner that is served by either unit's auxiliary component cooling water system.
   II. It appears that the same fans powered from Unit 2 motor control centers provide room cooling for both units' component cooling water pumps (potential documentation error). Note that room cooling requirements were screened out and, therefore, not included in the single unit PRA.
   III. The documentation does not describe if the units share the same circulating water bay or if they are separated.
d. There are multiple sequences in the utility's (and, therefore, the L3PRA project's) fire PRA that involve fires that start in Unit 2 and propagate to Unit 1. However, the "mechanics" of these fires and their propagation is not well understood due to gaps in the documentation of the utility's fire PRA available to the L3PRA project team.

D.3.2 Shared or Connected SSCs Between the SFPs and DCS and the Two Reactors

The DCS facility is a separate facility that does not have any shared or connected SSCs with the two reactors. Consequently, Table D-4 summarizes the combined results for shared or connected SSCs between only the SFPs and the two reactors for all hazards, both Level 1 and 2 PRAs, and FLEX strategies.

The results in Table D-4 are presented for two cases: (1) the main analysis, which addresses only scenarios that lead to fuel uncovery within 7 days, and (2) a sensitivity case that addresses scenarios that extend beyond the 7-day truncation time. Table D-4 contains the following columns:

- category (of potential sitewide dependency)
- identified dependencies
- relevant hazards and MUIEs
- notes
- keys inputs to modeling decisions

Section D.2.2 described the mitigative strategies for the SFPs in detail. Consequently, descriptions of these strategies are not repeated here. Based on the descriptions given in Section D.2.2 for the base case and sensitivity case analyses, the SFPs share the following systems and components with the two reactors:

- The NSCW systems (internal EDMG strategy – base case), including:
    - NSCW pumps
    - NSCW tower fans
- The fire protection system (specifically, standpipes and hoses) (internal EDMG strategy – base case)
- B.5.b pump (external EDMG strategy – base case)
- EDGs (sensitivity case)
- Valves needed to facilitate gravity makeup from the RWSTs, RMWSTs, or DWST (sensitivity case)

Similarly, the SFPs share the following structures with the two reactors for both the base case and sensitivity case:

- auxiliary building
- fuel handling building
- NSCW intake structure

In summary, Table D-4 shows that:

- For the base case, there are clear dependencies due to sharing of equipment and personnel via EDMG strategies.

- For the sensitivity case, there also are clear, though fewer, dependencies for the EDMG strategy that uses the SFPCPS.

- The SFPs share three structures with the two reactor units (i.e., auxiliary building, fuel handling building, and NSCW intake structure).

Table D-4 Shared or Connected SSCs Between the SFPs with the Reactors

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Keys Inputs to Modeling Decisions
Main Analysis (Base Case)				
Shared or connected systems and components	*EDMG Internal Makeup and Spray Strategy*:	Seismic events[a] (also applies to reactor-side LOINV during low power and shutdown [LPSD] when reactor is connected to the SFP)	EDMG internal and external makeup and spray strategies are the only mitigation strategies credited in the base case. A sensitivity analysis covering additional events considers additional strategies.	There is an obvious potential dependency between the SFPs and the reactors for equipment and personnel in EDMG strategies.
	NSCW system – pumps, tower fans, etc.			
	Fire protection system (fed from NSCW) – firewater standpipes, hoses, tie-downs, etc.			
	EDGs (to power NSCW pumps)			
	EDMG External Makeup and Spray Strategy: B.5.b pump[b] and trailer, hoses, etc.	Seismic events (also applies to reactor-side LOINV during LPSD when reactor is connected to the SFP)	EDMG internal and external makeup and spray strategies are the only mitigation strategies credited in the base case. A sensitivity analysis covering additional events considers additional strategies.	For the seismic bins likely to fail the SFP, and thus require mitigation, the external strategy is likely to fail and therefore does not reduce risk much for the SFP. As such, the external EDMG strategy may not be a good candidate for addressing multi-unit effects.

Table D-4 Shared or Connected SSCs Between the SFPs with the Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Keys Inputs to Modeling Decisions
Shared or connected structures	Auxiliary building	Seismic events (also applies to reactor-side LOINV during LPSD when reactor is connected to the SFP)	The SFPs are housed in the fuel handling building, which shares air space with the auxiliary building at the ground level elevation. The analysis assumes that seismic failure of either building will preclude the access needed to accomplish the EDMG mitigation strategies credited in the base case.	These connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	Fuel handling building	Seismic events (also applies to reactor-side LOINV during LPSD when reactor is connected to the SFP)		These connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	NSCW intake structure	Seismic events (also applies to reactor-side LOINV during LPSD when reactor is connected to the SFP)	The EDMG internal strategies require the NSCW systems to be working.	This represents another dependency between the SFPs and the reactors.
Sensitivity on 7-day Sequence Truncation[c]				
Shared or connected systems and components	NSCW system	Seismic events and non-seismic events	In the 7-day sequence truncation sensitivity, for seismic events, the analysis assumes that the SFPCPS is available if offsite power is not lost. For non-seismic events, loss of normal SFPCPS (from a variety of causes) is generally the initiating event. The NSCW system supports the SFPCPS for the SFPs and provides cooling for reactor systems.	
	Component cooling water system	Seismic events and non-seismic events	In the 7-day sequence truncation sensitivity, for seismic events, the	

Table D-4 Shared or Connected SSCs Between the SFPs with the Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Keys Inputs to Modeling Decisions
			analysis assumes that the SFPCPS is available if offsite power is not lost. For non-seismic events, loss of normal SFPCPS (from a variety of causes) is generally the initiating event. The component cooling water system cools the SFPCPS for the SFPs and the RHR heat exchangers for the reactors.	
	EDGs	Non-seismic events	Credited for supplying power to the SFPCPS for LOOP events.	
	Valves needed to facilitate gravity makeup from the RWSTs, RMWSTs, or DWST	Non-seismic events	Credited as a backup for the SFPCPS for non-seismic events.	
Shared or connected structures	Auxiliary building	Seismic events	Same treatment as main analysis (described above).	These connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).
	Fuel handling building	Seismic events		These connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations).

Table D-4 Shared or Connected SSCs Between the SFPs with the Reactors (cont.)

Category	Identified Dependencies	Relevant Hazards and MUIEs	Notes	Keys Inputs to Modeling Decisions

Notes:

a. The L3PRA project did not quantitatively address seismically induced fires, due to the ongoing nature of work in this area at the time of project initiation. Instead, modeling of seismically induced fires was identified as a candidate for future research. Nonetheless, the dependence on the fire protection system for the internal EDMG strategy should be noted.

b. There are two B.5.b pumps (with associated trailer and hoses): (1) in the warehouse and (2) at the fire training facility.

c. This is the sensitivity to consider events beyond the 7-day sequence truncation time used in the main analysis (base case). The results were that seismic events without leaks could significantly contribute to the calculated significant fuel uncovery frequency (SFUF). Non-seismic events were much lower frequency. It's possible that consideration of multi-unit effects might increase the contribution from non-seismic events, but it seems unlikely.

D.4 Results for the Assessment of Coupling Between the Two Reactors

Using the results discussed in Sections D.2 and D.3 , the reactor units can be assessed as being either “tightly” or “loosely” coupled.

From the EPRI (2021a) guidance on assessing the extent of “coupling” between two reactor units given in Section B.6.3, the following are characteristics of “coupling”:

- “Tightly coupled” reactors have:
    - shared support systems
    - shared front-line systems
    - inter-unit electrical dependencies
    - common or shared structures

- “Loosely coupled” reactors have:
    - limited (or no) shared systems
    - major structures that are separated and/or unconnected

In addition, EPRI (2021a) states that both “loosely coupled” and “tightly coupled” units typically have the following types of dependencies:

- shared or common offsite power connections
- shared ultimate heat sink or cooling source
- common component types
- shared accident resources (e.g., FLEX equipment)
- common physical location
- common emergency operating procedures, operator training, etc.
- common emergency operations center

Table D-5 summarizes the results of Section D.2 and Section D.3 with respect to the characteristics of “tight” coupling. This table also contains an additional potential dependency—shared physical resources. From Table D-5, it can be seen that the only instances of potential “tight” coupling between the reactors are for certain fire scenarios and for seismic events (which are addressed under the Phase 3 assessment of sitewide dependencies). Consequently, the two reactors are considered to be “loosely coupled” for all hazards except certain fire scenarios and seismic events (which are addressed in Phase 3).

Table D-5 Catalog of the Reference Site Features Associated with "Tightly Coupled" Reactors

Potential Dependencies	Yes or No?	Notes
Shared support systems	**No**	
Shared front-line systems	**No**	
Shared components	**No**	For Level 2 PRA, there are an adequate number of B.5.b. pumps to implement EDMG strategies for both reactors.
Inter-unit electrical dependencies	"**Yes**" for main switchyards and alternate switchyard	This dependency is common to both "tightly" and "loosely" coupled reactors.
Shared physical resources	**"Yes"** for FWSTs needed to implement EDMG strategies in Level 2 PRA.	Relevant for Level 2 PRA only; will need to account for this in multi-unit model.
Common or shared structures	Internal events and internal floods: "**No**" for the auxiliary buildings, control buildings, and turbine buildings.	Although these buildings are connected, equipment is not close by.
	Internal fires: "**Yes**" for certain scenarios.	Both units share auxiliary buildings, control buildings, fuel handling buildings, and turbine buildings. Additionally, there are other areas, such as low and high voltage yards containing equipment from both units. It should be pointed out that SSCs for redundant trains and trains from different units do not coexist in the same fire zone.
	Internal fires: "**Yes**" for fires that can cause dual-unit MCR abandonment.	These scenarios will need to be represented in MU risk models.
	Seismic events: probably "**Yes**" for all common and connected buildings.	This type of dependency will be addressed under Phase 3.

APPENDIX E
IDENTIFICATION OF CROSS-SOURCE COMMON CAUSE FAILURES

This appendix presents the results for Phase 3 sitewide dependency assessment, specifically for the category of identical components that can result in cross-source common cause failures (CCFs). As for other sitewide dependency assessments, this task was performed as part of the integrated site risk (ISR) task.

Appendix H presents the coupling factors used in the L3PRA's cross-source risk estimates, including those factors for cross-source CCFs.

E.1 Approach for Identifying Cross-Source CCFs

Section B.7.1 also describes the approach used to identify potential cross-source dependencies involving identical components (i.e., CCFs). Based on the approach described in Section B.7.1, identical components that are modeled in both reactor units were considered for modeling cross-unit CCFs. Two different types of CCFs were identified and documented for potential consideration in the ISR task:

- CCFs that are already modeled in the L3PRA project PRA models
- new CCFs involving identical components across the two reactor units (or between the reactors and the spent fuel pools [SFPs] or dry cask storage [DCS])

Note that this identification did not include CCFs for components not already included within the single reactor PRA models (which is a scope choice made for the L3PRA project). Section B.7.1 also states that it was expected that the majority of CCFs relevant to MU risk would be identified from the single unit, Level 1 internal events PRA model. However, it was recognized that additional CCF groups could be modeled in Level 1 PRAs for other hazards or in the Level 2 PRAs. Consequently, analysts were asked to document any risk significant CCF groups for other hazards (e.g., fire or seismic) even if these groups had already been identified from the Level 1 internal events PRA. Such information could be helpful in later screening for the ISR task.

As a reminder, Section B.7 also states that all potential dependencies identified in the Phase 3 assessment are:

- typically modeled by adjustments to basic event (BE) probabilities, rather than logic modeling
- difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether CCF groups should be expanded and what adjustment factor to use for an expanded group)
- difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the technical support center, etc.
- typically require modeling that is beyond the PRA state-of-the-art

Section E.2 presents the results for the identification of potential cross-unit CCFs for the reactors where an existing CCF is already modeled in the single reactor PRAs, but the group size may need to be expanded to include components in both reactors. Section E.3 presents the results for the identification of existing BEs for a single component failure in the single reactor PRAs for which a potential cross-unit CCF should now be considered. Section E.4 presents the results of the identification of potential CCFs for SFPs with the two reactors.

E.2 Results for Reactors: CCF Group Expansion

There are two cases for CCFs that are already modeled in the single unit base PRA: (1) the existing CCF group size is also appropriate for a multi-unit risk model, or (2) the existing CCF group size must be expanded for the multi-unit risk model.

For example, if there are CCFs already modeled for a system that is shared by both reactor units (e.g., CCFs of service water pumps)[66] and the success criteria is not changed when going from the single unit to multi-unit model, then no expansion of the common cause component group (CCCG) is needed. However, if the CCFs in the single unit PRA model are not in a shared system (e.g., emergency diesel generator [EDG] CCFs), the CCCG would need to be expanded to address the combined set of components in both reactor units and new CCF parameters would need to be estimated.

CCF group expansion results for existing CCFs corresponding with the following PRA hazards or types are given in the subsections below:

- Level 1 PRA for internal events (Section E.2.1)
- Level 1 PRA for internal floods and Level 2 PRAs for internal events and internal floods (Section E.2.2)
- Level 1 PRA for internal fires, seismic events, and wind-related events (Section E.2.3)
- Level 1 PRA – FLEX sensitivity case (Section E.2.4)

The following should be noted regarding the results provided below:

- Internal fire, seismic and wind-related (e.g., high wind [HWD] and tornado [TOR]) PRA models use the event trees from the internal event PRA and supplement them with additional modeling and event-specific boundary conditions, as needed. Consequently, no new CCF BEs are introduced in these models.
- When Phase 3 sitewide dependencies were assessed, Level 2 PRAs were not available for internal fires, seismic events, and wind-related events. Consequently, the Level 2 PRA results given below only involve internal events and internal floods.

E.2.1 Potential Expansion of Existing CCF Groups for Level 1 PRA for Internal Events

Table E-1 summarizes the results for the identification of CCFs modeled in the Level 1 PRA for internal events. These results are organized by system and show the identified component as

[66] Note, this is not the case for the service water pumps at the reference plant.

well as the component failure mode. The components and associated failure modes that are shown bolded are the most risk significant. Table E-2 (showing Fussell-Vesely [FV] importance results) and Table E-3 (showing risk achievement worth [RAW] results) are the basis for the summary in Table E-1. The following criteria for determining which components and associated failure modes are risk-significant were used:

- FV importance greater than 0.005
- RAW greater than 2

Focusing on the bolded, or most risk-significant, results in Table E-1 only, the following components and associated failure modes are candidates for multi-unit CCF group expansion:

- nuclear service cooling water (NSCW) pumps – failure to run

- switchyard reserve auxiliary transformer (RAT) breakers – failure to open

- EDGs – load sequencer failure; EDG failure to start or run; fuel oil transfer pump failure to start

- auxiliary feedwater (AFW) – motor-driven pump (MDP) failure to run

Also, it should be noted that a few potential cross-unit CCFs were identified as part of Phase 1 identification of multi-unit initiating events (MUIEs). In particular, the failures shown in Table E-1 of NSCW pumps and residual heat removal (RHR) hot and cold leg valves correspond with loss of NSCW and interfacing system loss of coolant accidents (ISLOCAs), respectively.

E.2.2 Potential Expansion of Existing CCF Groups for Level 1 PRA for Internal Floods and Level 2 PRAs for Internal Events and Internal Floods

Table E-4 summarizes the additional CCF events that are important for the Level 1 PRA for internal floods and the Level 2 PRAs for internal events and internal floods. Note that only two types of components need to be added for consideration of potential multi-unit CCFs. Otherwise, all the Level 1 PRA CCF events that were previously identified are also important to the Level 1 PRA for internal floods and Level 2 PRA results for internal events and internal floods.

Table E-5 provides the details behind Table E-4, showing the CCF events with respect to RAW importance values for these results. (No additional significant CCF BEs were identified with FV ≥ 0.005 for the Level 2 PRA significant release categories.)

Note that the CCF events identified in Table E-4 have much lower risk importance than many of the CCF events for the Level 1 PRA for internal events.

Table E-1   Level 1 PRA for Internal Events CCF List

System	Components (*ordered by risk importance*)
Nuclear service cooling water (NSCW)	**Pumps (FTR**, relays**)**, cooling tower (CT) spray valves (FTO, FTC, relays), pumps (FTS), pump motor-operated valves (MOVs), CT fans (FTS, FTR), relays, temperature switches
Switchyard	**Reserve auxiliary transformer (RAT) breakers (FTO)**
Emergency diesel generators (EDGs)	**Load sequencers, EDGs (FTR/FTS), fuel oil transfer pumps (FTS**, relays, FTR**)**, vent dampers, vent fans, running relays
Auxiliary feedwater (AFW)	**Pumps (FTR)**, pump check valves (suction and discharge), feedline check valves, control valves, minimum flow valves (transmitters)
Electrical	Battery chargers, inverters
Reactor protection system (RPS)	Rod cluster control assemblies (RCCAs), reactor trip breakers, bistables, analog process logic modules, UV drivers, solid state logic
Instrumentation and control (I&C)	Engineered safety features actuation system (ESFAS)
Emergency core cooling system (ECCS)	Residual heat removal (RHR) pumps (FTS, FTR), RHR pump discharge check valves, containment sump suction and check valves, containment sumps, safety injection (SI) pump minimum flow valves, refueling water storage tank (RWST) suction valves (FTC), high pressure recirculation (HPR) suction check valves, high pressure injection (HPI) and low pressure injection (LPI) cold leg (CL) suction check valves, SI pump suction from RHR pumps valves, normal charging valves (FTC), centrifugal charging pumps (CCPs) (FTS)
Auxiliary component cooling water (ACCW)	Pumps (FTR)

<u>Failure mode acronyms</u>
FTS – failure to start
FTR – failure to run
FTO – failure to open
FTC – failure to close

Table E-2 Components and Associated Failure Modes with Fussell-Vesely ≥ 0.005 – Internal Events Level 1 PRA

CCF Basic Event	Description	FV
1-ACP-CRB-CF-A205301	SWITCHYARD AC BREAKERS AA205 AND BA301 FAIL FROM COMMON CAUSE TO OPEN	1.90E-01
1-EPS-SEQ-CF-FOAB	SEQUENCERS FAIL FROM COMMON CAUSE TO OPERATE	1.34E-01
1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.05E-01
1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN	1.90E-02
1-AFW-PMP-CF-RUN	AFW PUMPS FAIL FROM COMMON CAUSE TO RUN (EXCLUDING DRIVER)	5.76E-03
1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START	5.35E-03
1-EPS-MDP-FS-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO START	5.13E-03

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA

CCF Basic Event	Description	RAW
1-SWS-MDP-CF-FR-ABCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABCD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABCF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ACDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ADEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-BCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-BCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-CDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ABDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-ACDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-SWS-MDP-CF-FR-BCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.15E+03
1-IE-SWS-MDP-CR-123456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	3.39E+03
1-DCP-BCH-FC-AAABBABB-CC	BATTERY CHARGER 1AD1CA, 1AD1CB, 1BD1CA AND 1BD1CB FAIL BY CCF - Quadruple CCF	8.68E+02
1-RPS-ROD-CF-RCCAS	CCF 10 OR MORE RCCAS FAIL TO DROP	7.91E+02
1-RPS-BME-CF-RTBAB	CCF RTB-A AND RTB-B (MECHANICAL)	6.79E+02
1-EPS-SEQ-CF-FOAB	SEQUENCERS FAIL FROM COMMON CAUSE TO OPERATE	6.16E+02
1-ACP-CRB-CF-A205301	SWITCHYARD AC BREAKERS AA205 AND BA301 FAIL FROM COMMON CAUSE TO OPEN	5.36E+02
1-DCP-BAT-CF-ALL	125 VDC BATTERIES FAIL FROM COMMON CAUSE	4.72E+02
1-SWS-MOV-CF-1668A69A	NSCW CT SPRAY VALVES HV1668A, 1669A FAIL FROM COMMON CAUSE TO OPEN	3.75E+02
1-ACP-INV-FC-AD11BD12-CC	INVERTERS 1AD1I11/1BD1I12 FAIL BY COMMON CAUSE	3.70E+02
1-AFW-PMP-CF-RUN	AFW PUMPS FAIL FROM COMMON CAUSE TO RUN (EXCLUDING DRIVER)	3.69E+02
1-SWS-RLY-FC-AX36869_-CC	CCF OF AX3 RELAYS FOR OPEN/CLOSE NSCW MOVS 1HV1668A/B & 1669A/B AFTER LOSP	3.46E+02

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

CCF Basic Event	Description	RAW
1-AFW-CKV-CC-010214__-CC	AFW PUMPS DISCHARGE LINE CVS 001, 002, 014 FAIL TO OPEN - CCF	3.41E+02
1-AFW-CKV-CC-331358__-CC	AFW PUMPS SUCTION CVS 033, 013, 058 FAIL TO OPEN - CCF	3.41E+02
1-AFW-CKV-CF-PDCV	PUMP DISCHARGE CHECK VALVES 001, 002, AND 014 FAIL FROM COMMON CAUSE	3.40E+02
1-AFW-CKV-CF-PSCV	PUMP SUCTION CHECK VALVES 033, 058, AND 013 FAIL FROM COMMON CAUSE	3.40E+02
1-AFW-CKV-CF-SGCV	SG CHECK VALVES 125, 126, 127, AND 128 FAIL FROM COMMON CAUSE	3.37E+02
1-AFW-SCV-CC-1131415_-CC	SG AFW FEED LINESTOP CVs 113	3.37E+02
1-AFW-SCV-CC-1161314_-CC	SG AFW FEED LINE STOP CVs 116 & 113 & 114 FAIL TO OPEN -CCF	3.37E+02
1-AFW-SCV-CC-1161315_-CC	SG AFW FEED LINE STOP CVs 116 & 113 & 115 FAIL TO OPEN -CCF	3.37E+02
1-AFW-SCV-CC-1161415_-CC	SG AFW FEED LINE STOP CVs 116 & 114 & 115 FAIL TO OPEN -CCF	3.37E+02
1-SWS-MDP-CF-FS-ABCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABCD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABCF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ACDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ADEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-BCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-BCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-CDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ABDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-ACDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MDP-CF-FS-BCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	3.36E+02
1-SWS-MOV-CF-116-ABCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-SWS-RLY-FC-162_1ALL-CC	RELAYS 162-1 ASSOC WITH OPENING OF HV-11600	3.36E+02
1-SWS-MOV-CF-116-ABCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-ABCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-ABCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-ABDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-ACDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-BCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.36E+02
1-SWS-MOV-CF-116-ABDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-ABDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-ABEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-ACDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-ACDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-ACEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-BCDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-BCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-MOV-CF-116-BCEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	3.35E+02
1-SWS-CTF-CF-S-ABCDEFGH	System Generated Event based upon Rasp CCF event: 1-SWS-FAN-CF-S	3.28E+02
1-SWS-CTF-CF-R-ABCDEFGH	System Generated Event based upon Rasp CCF event: 1-SWS-CTF-CF-R	3.27E+02
1-AFW-SCV-CC-16131415-CC	SG AFW FEED LINE STOP VS 116 & 113 & 114 & 115 FAIL TO OPEN -CCF	3.22E+02
1-AFW-SCV-CC-HICCF___-CC	HIGH ORDER CCF COMB. CAUSED AFWS FAIL-STOP CV FTO- AF FLOW DIST LINES	3.22E+02
1-SWS-RLY-FC-162_1X89-CC	RELAYS 162-1X FOR OPENING HV1668A /BAND 1669A /B AFTER LOSP FAILS -CCF	3.04E+02
1-SWS-RLY-FC-162_1PPS-CC	CCF OF NSCW PPS TDE RELAYES 162-1 - overall CCF for CCFG=6	3.01E+02
1-SWS-SWT-FC-TY16689B-CC	NSCW RETURN WTR TEMP SWITCHES TY1668B&1669B FAIL - CCF	2.85E+02
1-EPS-RLY-FC-RUN1234_-CC	DG RUNNING RELAYS 1234 FAILBY COMMON CAUSE	2.40E+02
1-EPS-CKV-CC-FXFERP__-CC	CCF OF CVS IN DG FUEL XFER PUMPS TRAINS TO OPEN (047, 044, 053,050)	2.39E+02
1-EPS-TFL-FC-XFERPSIG-CC	DG FUEL XFER PUMP SINGAL LT CCF	2.36E+02
1-IE-SWS-MDP-CR-12346	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.69E+02

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-IE-SWS-MDP-CR-12345	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.69E+02
1-RPS-CBI-CF-6OF8	CCF 6 BISTABLES IN 3 OF 4 CHANNELS	1.68E+02
1-RPS-CCX-CF-6OF8	CCF 6 ANALOG PROCESS LOGIC MODULES IN 3 OF 4 CHANNELS	1.67E+02
1-IE-SWS-MDP-CR-12356	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-12456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-13456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-23456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1234	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1236	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1245	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1256	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1346	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-1456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-2345	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-2356	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-IE-SWS-MDP-CR-3456	System Generated Event based upon Rasp CCF event: 1-IE-SWS-MDP-CF-	1.52E+02
1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START	1.46E+02
1-EPS-MDP-FS-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO START	1.46E+02
1-EPS-MDP-FR-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO RUN	1.46E+02
1-EPS-MOT-CF-START	DG ROOM VENT FANS FAIL FROM COMMON CAUSE TO START	1.46E+02
1-EPS-PND-CF-1205X	DG VENT DAMPERS FAIL FROM COMMON CAUSE	1.46E+02
1-EPS-MOT-CF-RUN	DG ROOM VENT FANS FAIL FROM COMMON CAUSE TO RUN	1.46E+02
1-SWS-CTF-CF-FS-ALL	4 OR MORE (ALL COMBINATIONS) NSCW FANS FAIL FROM COMMON CAUSE TO START	9.04E+01
1-SWS-CTF-CF-FR-ALL	4 OR MORE (ALL COMBINATIONS) NSCW FANS FAIL FROM COMMON CAUSE TO RUN	9.00E+01
1-SWS-MDP-CF-FR-ABC	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ABD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ABE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-SWS-MDP-CF-FR-ABF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ACD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ADE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ADF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-BCD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-BCE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-BCF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-CDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-CDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.58E+01
1-SWS-MDP-CF-FR-ABCE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.36E+01
1-SWS-MDP-CF-FR-ABDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.36E+01
1-SWS-MDP-CF-FR-ACDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.36E+01
1-SWS-MDP-CF-FR-BCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	7.36E+01
1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN	5.95E+01
1-ACP-INV-FC-A1B2____-CC	INVERTERS 1AD1I1/1BD1I2 FAIL BY COMMON CAUSE	5.42E+01
1-ACP-INV-FC-A1B2C3D4-CC	INVERTERS 1AD1I1/B2/C3/D4 FAIL BY COMMON CAUSE	5.33E+01
1-ACP-INV-FC-A1B2C3__-CC	INVERTERS 1AD1I1/B2/C3 FAIL BY COMMON CAUSE	5.33E+01
1-ACP-INV-FC-A1B2__D4-CC	INVERTERS 1AD1I1/B2/D4 FAIL BY COMMON CAUSE	5.33E+01
1-SWS-MDP-CF-FRL-12356	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FRL	4.98E+01
1-SWS-MDP-CF-FRL-12456	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FRL	4.98E+01
1-SWS-MDP-CF-FRL-13456	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FRL	4.98E+01
1-SWS-MDP-CF-FRL-23456	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FRL	4.98E+01
1-SWS-RLY-FC-AX46869_-CC	RELAYS AX4 FOR OPENING NSCW 1HV1668A/B & 1669A/B AFTER LOSP FAILS - CCF	4.40E+01
1-DCP-BCH-FC-AA__BABB-CC	BATTERY CHARGERS 1AD1CA, 1BD1CA, AND 1BD1CB FAIL - triple CCF	3.39E+01
1-DCP-BCH-FC-__ABBABB-CC	BATTERY CHARGERS 1AD1CB, 1BD1CA, AND 1BD1CB FAIL - triple CCF	3.38E+01
1-DCP-BCH-FC-AAABBA__-CC	BATTERY CHARGERS 1AD1CA, 1AD1CB, AND 1BD1CA FAIL - triple CCF	3.34E+01
1-DCP-BCH-FC-AAAB__BB-CC	BATTERY CHARGERS 1AD1CA, 1AD1CB, AND 1BD1CB FAIL - triple CCF	3.34E+01

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-DCP-BCH-FC-____BABB-CC	BATTERY CHARGERS 1BD1CA AND 1BD1CB FAIL - DOUBLE CCF	3.32E+01
1-DCP-BCH-FC-AAAB____-CC	BATTERY CHARGERS 1AD1CA AND 1AD1CB FAIL - DOUBLE CCF	3.27E+01
1-ESF-ACT-CF-__SAFACT-CC	COMMON CAUSE FAILURE OF ESFAS TRAIN A AND TRAIN B	3.21E+01
1-SWS-MDP-CF-FR-BD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.34E+01
1-SWS-MDP-CF-FR-BF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.34E+01
1-SWS-MDP-CF-FR-DF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.34E+01
1-SWS-MDP-CF-FR-AC	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.33E+01
1-SWS-MDP-CF-FR-AE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.33E+01
1-SWS-MDP-CF-FR-CE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	2.33E+01
1-AFW-MDP-CF-START	AFW MOTOR-DRIVEN PUMPS FAIL FROM COMMON CAUSE TO START	2.24E+01
1-AFW-MOV-CF-MINFL	AFW MDP MIN FLOW VALVES 5155 AND 5154 FAIL FROM COMMON CAUSE	2.19E+01
1-AFW-MDP-CF-RUN	AFW MOTOR-DRIVEN PUMPS FAIL FROM COMMON CAUSE TO RUN	2.14E+01
1-AFW-CKV-CC-001002__-CC	AFW PUMPS DISCHARGE LINE CVS 001, 002 FAIL TO OPEN - CCFs	1.81E+01
1-AFW-CKV-CC-033058__-CC	AFW PUMPS SUCTION CVS 033, 058 FAIL TO OPEN -CCF	1.81E+01
1-SWS-MDP-CF-FR-ACE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-ACF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-AEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-BDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-BDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-BEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-CEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-DEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.72E+01
1-SWS-MDP-CF-FR-ACEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.66E+01
1-SWS-MDP-CF-FR-BDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FR	1.66E+01
1-LPI-MDP-CF-START	RHR PUMPS A, B FAIL FROM COMMON CAUSE TO START	1.62E+01
1-SWS-MDP-CF-FS-ABD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01
1-SWS-MDP-CF-FS-ABF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-SWS-MDP-CF-FS-ADF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01
1-SWS-MDP-CF-FS-BCD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01
1-SWS-MDP-CF-FS-BCF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01
1-SWS-MDP-CF-FS-CDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.60E+01
1-AFW-TFF-CF-MINFL	AFW MINFLOW LINE FLOW TRANSMITERS FT-5155 AND FT-5154 FAIL FROM COMMON CAUSE	1.60E+01
1-SWS-MDP-CF-FS-ABC	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-SWS-MDP-CF-FS-ABE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-SWS-MDP-CF-FS-ACD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-SWS-MDP-CF-FS-ADE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-SWS-MDP-CF-FS-BCE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-SWS-MDP-CF-FS-CDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.59E+01
1-LPI-MDP-CF-RUN	RHR PUMPS A, B FAIL FROM COMMON CAUSE TO RUN	1.58E+01
1-SWS-MDP-CF-FS-ABDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.57E+01
1-SWS-MDP-CF-FS-BCDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.57E+01
1-SWS-MDP-CF-FS-ABCE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.55E+01
1-SWS-MDP-CF-FS-ACDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.55E+01
1-LPI-CKV-CC-009010__-CC	RHR Pumps Discharge CVs 009 AND 010 FAIL TO OPEN BY COMMON CAUSE	1.54E+01
1-LPI-CKV-CF-009010	RHR PUMP DISCHARGE CVs 009, 010 FAIL FROM COMMON CAUSE TO OPEN	1.53E+01
1-SWS-MDP-CF-FS-BDE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.51E+01
1-SWS-MDP-CF-FS-BEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.51E+01
1-SWS-MDP-CF-FS-DEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.51E+01
1-SWS-MDP-CF-FS-ACF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.49E+01
1-SWS-MDP-CF-FS-AEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.49E+01
1-SWS-MDP-CF-FS-CEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.49E+01
1-SWS-MDP-CF-FS-BDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.48E+01
1-SWS-MDP-CF-FS-ACEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.46E+01
1-SWS-MOV-CF-116-ADE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-SWS-MOV-CF-116-ADF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01
1-SWS-MOV-CF-116-AEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01
1-SWS-MOV-CF-116-CDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01
1-SWS-MOV-CF-116-CDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01
1-SWS-MOV-CF-116-CEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.45E+01
1-SWS-MDP-CF-FS-BD	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.44E+01
1-SWS-MDP-CF-FS-BF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.44E+01
1-SWS-MDP-CF-FS-DF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.44E+01
1-SWS-MOV-CF-116-ABD	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MOV-CF-116-ABF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MOV-CF-116-ACD	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MOV-CF-116-ACF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MOV-CF-116-BCD	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MOV-CF-116-BCF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.43E+01
1-SWS-MDP-CF-FS-AC	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.43E+01
1-SWS-MDP-CF-FS-AE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.43E+01
1-SWS-MDP-CF-FS-CE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.43E+01
1-SWS-MDP-CF-FS-BDF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.39E+01
1-SWS-MDP-CF-FS-ACE	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1.38E+01
1-SWS-MOV-CF-116-BDE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.36E+01
1-SWS-MOV-CF-116-BDF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.36E+01
1-SWS-MOV-CF-116-BEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.36E+01
1-SWS-MOV-CF-116-ADEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.36E+01
1-SWS-MOV-CF-116-CDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.36E+01
1-SWS-MOV-CF-116-ABE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.35E+01
1-SWS-MOV-CF-116-ACE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.35E+01
1-SWS-MOV-CF-116-BCE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.35E+01

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-SWS-MOV-CF-116-ABCD	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.35E+01
1-SWS-MOV-CF-116-ABCF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.35E+01
1-SWS-MOV-CF-116-DE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.32E+01
1-SWS-MOV-CF-116-DF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.32E+01
1-SWS-MOV-CF-116-EF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.32E+01
1-SWS-MOV-CF-116-AB	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.30E+01
1-SWS-MOV-CF-116-AC	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.30E+01
1-SWS-MOV-CF-116-BC	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.30E+01
1-SWS-MOV-CF-116-DEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.29E+01
1-SWS-MOV-CF-116-BDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.29E+01
1-SWS-MOV-CF-116-ABCE	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.27E+01
1-SWS-MOV-CF-116-ABC	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1.27E+01
1-EPS-RLY-FC-RUN24___-CC	DG RUNNING RELAYS 24 FAIL BY COMMON CAUSE	1.16E+01
1-EPS-RLY-FC-RUN13___-CC	DG RUNNING RELAYS 13 FAIL BY COMMON CAUSE	1.15E+01
1-RPS-CBI-CF-4OF6	CCF 4 BISTABLES IN 2 OF 3 CHANNELS	1.12E+01
1-RPS-CCX-CF-4OF6	CCF 4 ANALOG PROCESS LOGIC MODULES IN 2 OF 3 CHANNELS	1.11E+01
1-LPI-MOV-CF-8811AB	RHR CONTAINMENT SUMP SUCTION MOVs HV8811A & B FAIL FROM COMMON CAUSE TO OPEN	9.31E+00
1-HPI-MOV-CF-8804AB	HV8804A, HV8804B FAIL FROM COMMON CAUSE TO OPEN	9.28E+00
1-LPI-MOV-CF-8812AB	RWST SUCTION MOVSs HV8812A & B FAIL FROM COMMON CAUSE TO CLOSE	9.25E+00
1-HPI-MOV-OO-13148920-CC	SI PUMPS MINI FLOW ISOLATION MOVS HV8813 & 8814 & 8920 FAILS TO CLOSE - CCF	9.25E+00
1-HPI-MOV-OO-88138814-CC	SI PUMPS MINI FLOW ISOLATION MOVS HV8813 & 8814 FAILS TO CLOSE - CCF	9.21E+00
1-HPI-MOV-OO-88138920-CC	SI PUMPS MINI FLOW ISOLATION MOVS HV8813 & 8920 FAILS TO CLOSE - CCF	9.21E+00
1-LPI-CKV-CC-122123__-CC	CONTAINMENT SUMP CVs 122 and 123 (RHRP suction) FAIL TO OPEN BY COMMON CAUSE	9.15E+00
1-HPI-CKV-CF-436_163	HP RECIR SUCTION FROM RHR HXs CVs 436 & 163 FAIL FROM COMMON CAUSE TO OPEN	9.03E+00
1-LPI-CKV-CF-122123	CONTAINMENT SUMP CVs 122 and 123 (RHRP suction) FAIL FROM COMMON CAUSE TO OPEN	9.03E+00
1-LPI-SMP-CF-SUMPAB	ECCS CONTAINMENT SUMPS A, B FAIL FROM COMMON CAUSE PLUGGING	8.98E+00
1-EPS-RLY-FC-RUN123__-CC	DG RUNNING RELAYS 123 FAIL BY COMMON CAUSE	8.03E+00

Table E-3 Components and Associated Failure Modes with Risk Achievement Worth ≥ 2 – Internal Events Level 1 PRA (cont.)

1-EPS-RLY-FC-RUN124__-CC	DG RUNNING RELAYS 124 FAIL BY COMMON CAUSE	8.03E+00
1-EPS-RLY-FC-RUN134__-CC	DG RUNNING RELAYS 134 FAIL BY COMMON CAUSE	8.03E+00
1-EPS-RLY-FC-RUN234__-CC	DG RUNNING RELAYS 234 FAIL BY COMMON CAUSE	8.03E+00
1-AFW-CKV-CC-001014__-CC	AFW PUMP DISCHARGE LINE CVS 001, 014 FAIL TO OPEN -CCF	7.24E+00
1-AFW-CKV-CC-002014__-CC	AFW PUMPS DISCAHRGE CVS 002, 014 FAIL DUE - CCF	7.24E+00
1-AFW-CKV-CC-033013__-CC	AFW PUMPS SUCTION CVs 033, 013 FAIL TO OPEN -CCF	7.24E+00
1-AFW-CKV-CC-058013__-CC	AFW PUMPS SUCTION CVs 058, 013 FAIL TO OPEN - CCF	7.24E+00
1-HPI-CKV-CF-CLALL	COLD LEG CVs 083, 084, 085, 086 FAIL FROM COMMON CAUSE TO OPEN	7.01E+00
1-LPI-CKV-CF-CLALL	RHR COLD LEG CHECK VALVES 147, 148, 149, 150 FAIL FROM COMMON CAUSE TO OPEN	7.01E+00
1-IE-ACW-MDP-CF-FR12	CCF TO RUN OF ACCW PUMPS 1-1217-P4-001 & 002 - 1 YEAR EXPOSURE TIME	5.00E+00
1-ACW-MDP-CF-FR0012	CCF TO RUN OF ACCW PUMPS 1-1217-P4-001 AND 002 (24 HR)	4.60E+00
1-SWS-MOV-OO-1668A69A-CC	NSCW CT SPRAY VALVES HV1668A & 69A FAILS TO CLOSE DUE TO CCF	4.06E+00
1-AFW-SCV-CC-113114__-CC	SG AFW FEED LINE STOP CVs 113 & 114 FAIL TO OPEN - CCF	2.81E+00
1-AFW-SCV-CC-113115__-CC	SG AFW FEED LINE STOP CVs 113 & 115 FAIL TO OPEN - CCF	2.81E+00
1-AFW-SCV-CC-114115__-CC	SG AFW FEED LINE STOP CVs 114 & 115 FAIL TO OPEN -CCF	2.81E+00
1-AFW-SCV-CC-116114__-CC	SG FEED LINE STOP CVs 116 & 114 FAIL TO OPEN - CCF	2.81E+00
1-AFW-SCV-CC-116115__-CC	SG AFW FEED LINE STOP CVs 116 & 115 FAIL TO OPEN - CCF	2.81E+00
1-HPI-MOV-OO-HV8105&6-CC	Normal Charging Isolation MOVs HV8106 & HV8105 FAIL TO CLOSE due to CCF	2.39E+00
1-RPS-UVL-CF-UVDAB	CCF UV DRIVERS TRAINS A AND B (2 OF 2)	2.39E+00
1-RPS-TLC-CF-SSLAB	CCF SOLID STATE LOGIC IN TRAINS A AND B (4 OF 4)	2.16E+00
1-CVC-MDP-FR-CCPACCPB-CC	CCP-A AND CCP-B FAIL TO RUN DUE TO COMMON CAUSE	2.04E+00

Table E-4 Level 2 PRA for Internal Events CCF List

System	Components (*ordered by risk importance*)
ECCS sump	ECCS sumps fail due to plugging
Containment isolation	Containment isolation valves fail to operate

Table E-5 Level 2 PRA Results for RAW ≥ 2

CCF Basic Event	Description	RAW
1-LPI-SMP-CF-SUMPAB	CCF OF ECCS CONTAINMENT SUMPS A & B FROM PLUGGING	6.69E+00
1-CIS-AOV-OO-2626_27B-CC	AOV HV-2626B & AOV HV-2627B FAIL TO OPERATE (HARDWARE)	2.07E+00
1-CIS-AOV-OO-HV28_29B-CC	AOV HV-2628B & AOV HV-2629B FAIL TO OPERATE (HARDWARE)	2.07E+00
1-CIS-AOV-OO-HV780781-CC	AOV HV-0780 & AOV HV-0781 FAIL TO OPERATE (HARDWARE)	2.07E+00

E.2.3 Potential Expansion of Existing CCF Groups for Level 1 PRA for Internal Fires, Seismic Events, and Wind-Related Events

For fire, seismic, and wind (F/S/W) events, component importance tables were created from SAPHIRE and CCF BEs were identified and separated out. These CCF events are sorted by their FV and risk increase ratio (RIR) importances.

The CCF BEs with the highest component importances by their FV values are shown in Table E-6, Table E-7, and Table E-8 for fires, wind-related events, and seismic events, respectively. Colors (e.g., green or blue) are used in Table E-6, Table E-7, and Table E-8 to draw attention to the column of FV results.

Highlights from Table E-6, Table E-7, and Table E-8 include:

- For internal fires, CCF events from various systems show up at the top of the sorted FV list. This is expected since quite a few fire scenarios (210) at different plant locations are modeled, with fire damage postulated for different types of system equipment, resulting in a broad spectrum of complementary systems (to avoid core damage) to be risk significant.

- An examination of the CCF BEs for seismic and wind shows that the most significant CCF failures are related to those that also show up for the LOOP events.

- RPS CCF failures show up as significant for fire and seismic events.

The following systems (and components/failure modes) are examples of results shown in Table E-6, Table E-7, and Table E-8 (which appear to overlap some of the Level 1 PRA results for internal events):

- Switchyard – AC CRBs
- EDGs – fail to start, fail to run, fuel transfer pumps fail to start
- AFW – MDPs (fail to start, fail to run)
- RPS – RCCAs, bi-stables
- ESFAS – fail to actuate
- RHR – pumps fail to run
- NSCW – containment spray valves fail to open or close

E.2.4 Potential Expansion of Existing CCF Groups for Level 1 PRA - FLEX Sensitivity Case

The portable FLEX equipment stored on site for both units is listed in the reference site Final Integrated Plan (reproduced below as Table E-9). The equipment satisfies the “N+1” requirement (see below) for the site with two units. Thus, each type of equipment listed is redundant and any one of them may be used for either Unit 1 or Unit 2. It is assumed that, for a given type of equipment, all redundant components are identical.

The FLEX sensitivity analysis for the L3PRA CDF calculations does not individually model the portable FLEX equipment shown in Table E-9. There are no new CCF BEs introduced by the FLEX sensitivity analysis model. This applies equally to all six hazard categories modeled.

For each type of portable equipment shown in Table E-9, where the quantity is more than one, there is CCF potential. If the quantity of equipment is three or greater, then a CCF BE could be modeled for a single unit and a cross-unit CCF could be modeled for both units. Although CCFs could be modeled such that the CCF group size is the same as the quantity shown in the table, CCF modeling should be consistent with how FLEX strategies are expected to be implemented (e.g., each unit is assigned a specific FLEX diesel generator [DG] and FLEX pump; any additional FLEX DGs or FLEX pumps can be used by either unit, as needed).

Examples of FLEX equipment from Table E-9 that could be modeled with single unit CCF BEs (i.e., N+1 or greater) and cross-unit CCFs include:

- 480 V FLEX DGs
- steam generator (SG) FLEX pumps
- boron injection FLEX pumps
- reactor coolant system (RCS) makeup FLEX pumps
- FLEX fuel tankers

Examples of FLEX equipment from Table E-9 that could be modeled with cross-unit CCFs only include:

- tow vehicles
- makeup FLEX pumps

Also, the reference plant Final Integrated Plan states that the “N+1” requirement does not apply to the FLEX support equipment, vehicles, and tools. However, these items are subject to inventory checks, requirements, and any associated maintenance and testing.

Table E-6 CCFs in Internal Fire PRA (CD-FRI CCF BEs)

Index #	Name	Prob	FV	RIR	Description
96	1-ACP-CRB-CF-A205301	3.50E-04	3.72E-02	1.07E+02	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
585	1-EPS-DGN-CF-FRUN1	3.24E-04	8.82E-03	2.82E+01	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
1286	1-RPS-BME-CF-RTBAB	1.61E-06	8.36E-03	4.49E+03	CCF OF RTB-A & RTB-B (MECHANICAL)
326	1-AFW-PMP-CF-RUN	1.55E-05	7.65E-03	4.88E+02	CCF OF AFW PUMPS TO RUN (EXCLUDING DRIVER)
1318	1-RPS-ROD-CF-RCCAS	1.21E-06	6.12E-03	4.38E+03	CCF 10 OR MORE RCCAS FAIL TO DROP
1294	1-RPS-CBI-CF-6OF8	2.70E-06	2.78E-03	1.00E+03	CCF OF 6 BISTABLES IN 3 OF 4 CHANNELS
641	1-ESF-ACT-CF-__SAFACT-CC	6.83E-05	2.28E-03	3.44E+01	CCF OF ESFAS TRAIN A & TRAIN B
1045	1-LPI-MDP-CF-START	4.88E-05	2.20E-03	4.61E+01	CCF OF RHR PUMPS A & B TO START
1303	1-RPS-CCX-CF-6OF8	1.83E-06	1.88E-03	1.00E+03	CCF OF 6 ANALOG PROCESS LOGIC MODULES IN 3 OF 4 CHANNELS
290	1-AFW-MDP-CF-START	5.02E-05	1.58E-03	3.24E+01	CCF OF AFW MDPs TO START
586	1-EPS-DGN-CF-FSUN1	3.68E-05	1.47E-03	4.10E+01	CCF OF UNIT 1 DGNs G4001/G4002 TO START
602	1-EPS-MDP-FS-XFERPPS_-CC	3.53E-05	1.43E-03	4.15E+01	CCF OF DG FUEL TRANSFER PUMPS TO START
1685	1-SWS-MOV-CF-1668A69A	1.19E-05	9.83E-04	8.36E+01	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN
1726	1-SWS-SWT-FC-TY16689B-CC	1.17E-05	9.56E-04	8.25E+01	CCF OF NSCW RETURN WATER TEMP SWITCHES TY1668B&1669B
629	1-EPS-SEQ-CF-FOAB	2.15E-04	8.37E-04	4.90E+00	CCF OF SEQUENCERS TO OPERATE
1375	1-SWS-CTF-CF-FS-ALL	1.05E-05	7.96E-04	7.68E+01	CCF OF 4 OR MORE (ALL COMBINATIONS) NSCW FANS TO START
1293	1-RPS-CBI-CF-4OF6	8.21E-06	5.19E-04	6.42E+01	CCF OF 4 BISTABLES IN 2 OF 3 CHANNELS
1302	1-RPS-CCX-CF-4OF6	6.33E-06	4.00E-04	6.41E+01	CCF OF 4 ANALOG PROCESS LOGIC MODULES IN 2 OF 3 CHANNELS
498	1-DCP-BCH-FC-AAABBABB-CC	1.53E-06	3.66E-04	2.39E+02	CCF OF BCHs 1AD1CA, 1AD1CB, 1BD1CA, & 1BD1CB
307	1-AFW-MOV-CF-MINFL	1.06E-05	3.25E-04	3.18E+01	CCF OF AFW MDP MINI FLOW VALVES 5155 & 5154
439	1-CCU-MOT-FS-CCUALL__-CC	2.13E-04	3.08E-04	2.45E+00	HIGH ORDER CCF COMB CAUSING CCU SYSTEM FAILURE TO START
597	1-EPS-MDP-FR-XFERPPS_-CC	7.26E-06	2.93E-04	4.13E+01	CCF OF DG FUEL TRANSFER PUMPS TO RUN
1565	1-SWS-MDP-CF-FS-ABCDEF	4.21E-06	2.87E-04	6.89E+01	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table E-6   CCFs in Internal Fires PRA (CD-FRI CCF BEs) (cont.)

Index #	Name	Prob	FV	RIR	Description
1259	1-RCS-PRV-CF-RV5A6A__	1.04E-04	2.68E-04	3.57E+00	CCF OF PORVS PV0455A (5A) & PV0456A (6A) TO OPEN
1517	1-SWS-MDP-CF-FR-ABCDEF	8.36E-08	2.18E-04	2.41E+03	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FR
1070	1-LPI-MOV-CF-8811AB	1.19E-05	2.01E-04	1.79E+01	CCF OF RHR CONTAINMENT SUMP SUCTION MOVs HV8811A & B TO OPEN
1095	1-MSS-ADV-CC-VPV0123_-CC	4.45E-05	1.98E-04	5.44E+00	CCF OF SG ARVS PV-3000, PV-3010, PV-3020, & PV-3030 TO OPEN
289	1-AFW-MDP-CF-RUN	6.07E-06	1.85E-04	3.14E+01	CCF OF AFW MDPs TO RUN
1044	1-LPI-MDP-CF-RUN	3.94E-06	1.74E-04	4.52E+01	CCF OF RHR PUMPS A & B TO RUN
1690	1-SWS-MOV-OO-1668A69A-CC	2.48E-04	1.52E-04	1.61E+00	CCF OF NSCW CT SPRAY VALVES HV1668A & 69A TO CLOSE
171	1-ACP-INV-FC-AD11BD12-CC	1.21E-06	1.39E-04	1.16E+02	CCF OF INVERTERS 1AD1I11/1BD1I12
1563	1-SWS-MDP-CF-FS-ABCD	2.00E-06	1.35E-04	6.84E+01	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table E-7 CCFs in Wind-Related PRA (CD-HWD and CD-TOR BEs)

Index #	Name	Prob	FV	RIR	Description
47	1-ACP-CRB-CF-A205301	3.50E-04	9.07E-02	2.00E+02	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
208	1-EPS-DGN-CF-FRUN1	3.24E-04	5.71E-02	1.26E+02	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
238	1-EPS-SEQ-CF-FOAB	2.15E-04	5.57E-02	2.00E+02	CCF OF SEQUENCERS TO OPERATE
449	1-OEP-XHE-XX-NR02HWR0	4.86E-01	1.07E-02	1.01E+00	CONVOLUTION FACTOR FOR CCF-OPR (2HR-WR Avail)
209	1-EPS-DGN-CF-FSUN1	3.68E-05	7.86E-03	1.60E+02	CCF OF UNIT 1 DGNs G4001/G4002 TO START
221	1-EPS-MDP-FS-XFERPPS_-CC	3.53E-05	7.54E-03	1.60E+02	CCF OF DG FUEL TRANSFER PUMPS TO START
681	1-SWS-MOV-CF-1668A69A	1.19E-05	2.89E-03	1.84E+02	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN
216	1-EPS-MDP-FR-XFERPPS_-CC	7.26E-06	1.55E-03	1.60E+02	CCF OF DG FUEL TRANSFER PUMPS TO RUN
570	1-SWS-MDP-CF-FS-ABCDEF	4.21E-06	1.02E-03	1.84E+02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
443	1-OEP-XHE-XX-NR01HWR0	4.35E-01	5.99E-04	1.00E+00	CONVOLUTION FACTOR FOR CCF-OPR (1HR-WR Avail)
223	1-EPS-MOT-CF-START	2.80E-06	5.97E-04	1.59E+02	CCF OF DG ROOM VENT FANS FAIL FROM COMMON CAUSE TO START
686	1-SWS-MOV-OO-1668A69A-CC	2.48E-04	5.42E-04	3.17E+00	CCF OF NSCW CT SPRAY VALVES HV1668A & 69A TO CLOSE
568	1-SWS-MDP-CF-FS-ABCD	2.00E-06	4.83E-04	1.84E+02	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table E-8 CCFs in Seismic PRA (CD-EQ-CCF BEs)

Index #	Name	Prob	FV	RIR	Description
125	1-ACP-CRB-CF-A205301	3.50E-04	1.00E-02	2.97E+01	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
464	1-EPS-DGN-CF-FRUN1	3.24E-04	9.28E-03	2.97E+01	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
535	1-ESF-ACT-CF-__SAFACT-CC	6.83E-05	6.83E-03	1.01E+02	CCF OF ESFAS TRAIN A & TRAIN B
520	1-EPS-SEQ-CF-FOAB	2.15E-04	6.17E-03	2.97E+01	CCF OF SEQUENCERS TO OPERATE
465	1-EPS-DGN-CF-FSUN1	3.68E-05	1.05E-03	2.97E+01	CCF OF UNIT 1 DGNs G4001/G4002 TO START
834	1-RPS-UVL-CF-UVDAB	1.04E-05	1.04E-03	1.01E+02	CCF OF UV DRIVERS TRAINS A & B (2 OF 2)
501	1-EPS-MDP-FS-XFERPPS_-CC	3.53E-05	1.01E-03	2.97E+01	CCF OF DG FUEL TRANSFER PUMPS TO START
272	1-AFW-PMP-CF-RUN	1.55E-05	4.31E-04	2.88E+01	CCF OF AFW PUMPS TO RUN (EXCLUDING DRIVER)
1038	1-SWS-MOV-CF-1668A69A	1.19E-05	3.80E-04	3.30E+01	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN
790	1-RPS-CBI-CF-6OF8	2.70E-06	2.53E-04	9.48E+01	CCF OF 6 BISTABLES IN 3 OF 4 CHANNELS
830	1-RPS-TLC-CF-SSLAB	2.10E-06	2.09E-04	1.01E+02	CCF OF SOLID STATE LOGIC IN TRAINS A & B (4 OF 4)
496	1-EPS-MDP-FR-XFERPPS_-CC	7.26E-06	2.08E-04	2.97E+01	CCF OF DG FUEL TRANSFER PUMPS TO RUN
793	1-RPS-CCX-CF-6OF8	1.83E-06	1.72E-04	9.48E+01	CCF OF 6 ANALOG PROCESS LOGIC MODULES IN 3 OF 4 CHANNELS
784	1-RPS-BME-CF-RTBAB	1.61E-06	1.60E-04	1.01E+02	CCF OF RTB-A & RTB-B (MECHANICAL)
796	1-RPS-ROD-CF-RCCAS	1.21E-06	1.55E-04	1.29E+02	CCF 10 OR MORE RCCAS FAIL TO DROP
921	1-SWS-MDP-CF-FS-ABCDEF	4.21E-06	1.21E-04	2.97E+01	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS
170	1-ACP-INV-FC-A1B2____-CC	1.21E-06	1.21E-04	1.01E+02	CCF OF INVERTERS 1AD1I1/1BD1I2
255	1-AFW-MDP-CF-START	5.02E-05	9.17E-05	2.83E+00	CCF OF AFW MDPs TO START
503	1-EPS-MOT-CF-START	2.80E-06	8.02E-05	2.96E+01	CCF OF DG ROOM VENT FANS FAIL FROM COMMON CAUSE TO START
1060	1-SWS-SWT-FC-TY16689B-CC	1.17E-05	7.14E-05	7.10E+00	CCF OF NSCW RETURN WATER TEMP SWITCHES TY1668B&1669B
1043	1-SWS-MOV-OO-1668A69A-CC	2.48E-04	6.42E-05	1.26E+00	CCF OF NSCW CT SPRAY VALVES HV1668A & 69A TO CLOSE
617	1-LPI-MDP-CF-START	4.88E-05	5.91E-05	2.21E+00	CCF OF RHR PUMPS A & B TO START
919	1-SWS-MDP-CF-FS-ABCD	2.00E-06	5.73E-05	2.96E+01	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS

Table E-9 PWR Portable Equipment Stored On-Site

Use and (Potential / Flexibility) Diverse Uses							Performance Criteria
List Portable Equipment	Qty	Core	Containment	SFP	Instrumentation	Accessibility	
Medium-Wheeled Loader -Can also be used as a towvehicle	1	X	X	X	X	X	Debris Removal
Tow Vehicles - 1 large, 1 small	2	X	X	X	X	X	Towing Pumps and Diesel Generators
480V FLEX Diesel Generator	3	X			X		Provide 480V AC power to FLEX Switchboard
SG FLEX Pump	3	X					Provides injection into the SGs to removedecay heat from the core.
Makeup FLEX Pump	2	X					Provide CST Makeup - Godwin HL11OM
Makeup FLEX Pump	1	X					Provide CST Makeup - Godwin HL-4M
SFP FLEX Submersible Pump Hydraulic Unit	2	X		X			Provides the hydraulic motive force to drivethe submersible pump
SFP FLEX Pump Submersible Pumps	4	X		X			Pump unit placed in the NSCW Basin for access to entire water volume
Sets of Monitor Spray Nozzles for SFP Spray and required hoses	6			X			Provides 250 gpm of spray water for each unit
Boron Injection FLEX Pump	3	X					Provides Borated Water from the BAST or RWST for injection to the RCS in MODES with SGs available for decay heat removal
RCS Makeup FLEX Pump	3	X					Provides borated water from the RWST for injection to the RCS during MODES with SGsnot available for decay heat removal

Table E-9 PWR Portable Equipment Stored On-Site (cont.)

	Use and (Potential / Flexibility) Diverse Uses						Performance Criteria
List Portable Equipment	Qty	Core	Containment	SFP	Instrumentation	Accessibility	
FLEX Fuel Tanker	3	X	X	X	X		Provide fuel to diesel powered FLEX equipment.
20 kW FLEX Diesel Generator	3						Not credited in FLEX strategies
DC Equipment Room FLEX Fan	10	X	X	X	X	X	Not credited in FLEX strategies. Portableventilation for equipment operability.
Battery Room FLEX Fan	10						Not credited in FLEX strategies. Portable ventilation fans available for long term cooling.
FLEX Ventilation Fan	2	X	X	X	X	X	For MCR Ventilation
Diesel Powered Lights	4						Misc. lighting. Not credited in FLEX strategies
Air Compressors	2						Air as needed. Not credited in FLEX strategies
Rapidly Deployable Communications Kit	2	X	X	X	X	X	Does not rely on the availability of either on-site or off-site infrastructure other than satellites

E.3 <u>Results for Reactors: Single Component Failures to Consider as New Multi-Unit CCFs</u>

This section addresses the identification and consideration of new potential multi-unit CCFs (MUCCFs) for inclusion in the multi-unit risk model. Specifically, this identification focuses on those BEs (component failures) where one such component exists in each unit, so that CCF of this component type is not considered in the single unit PRA but may be appropriate for the multi-unit PRA.

Parallel to Section E.2, single component failures to consider as new MUCCFs are addressed in the subsections below grouped by the following PRA hazards or types:

- Level 1 PRA for internal events (Section E.3.1)
- Level 1 PRA for internal floods and Level 2 PRAs for internal events and internal floods (Section E.3.2)
- Level 1 PRA for internal fires, seismic events, and wind-related events (Section E.3.3)
- Level 1 PRA – FLEX sensitivity case (Section E.3.4)

E.3.1 Single Component Failures to Consider as MUCCFs from Level 1 PRA for Internal Events

A review of the Level 1 PRA for internal events and its results led to the identification of the single components shown in Table E-10 as potential candidates for cross-unit CCFs. As for Table E-10, the results shown in Table E-10 are ranked by risk importance with the most risk significant components and associated failure modes shown in bold text. The results shown in Table E-10 are based on Table E-11 (single component failures with FV importance of 0.005 or greater) and Table E-12 (single component failures with RAW of 2 or greater).

Focusing on the bolded results in Table E-10 only, the following single components and associated failure modes are candidates for a new cross-unit CCF:

- AFW – turbine-driven pump fails to run
- ECCS – Normal charging pump (NCP) fails to run

Table E-10 Level 1 PRA for Internal Events – Single Component Failures for Multi-Unit CCF Consideration (Ordered by Risk Importance)

System	Components (*ordered by risk importance*)
AFW	**Turbine-driven pump (TDP) (FTR)**
ECCS	**NCP (FTR)**, refueling water storage tank (RWST); RWST suction valve
ACCW	Surge tank

FTR – failure to run
FTS – failure to start

Table E-11 Single Component Failures in Level 1 Internal Events PRA – FV ≥ 0.005

CCF Basic Event	Description	FV
1-CVC-MDP-FR-NCP4001&	NORMAL CHARGING PUMP 1208P4001 FAILS TO RUN (1 YEAR)	1.63E-02
1-AFW-TDP-FR-P4001___	TDAFWP (P4-001) FAILS TO RUN	1.34E-02

Table E-12 Single Component Failures in Level 1 Internal Events PRA – RAW ≥ 2

CCF Basic Event	Description	RAW
1-HPI-TNK-RP-RWST____	TANK RUPTURES	1.57E+01
1-HPI-XVM-PG-207_____	MANUAL VALVE 207 PLUGS	1.54E+01
1-IE-ACW-TNK-RP-T4_001_	ACCW SURGE TANK 1-1217-T4-001 RUPTURES CAUSINGLOW LOW LEVEL - ONE YEAR EXPOSURE	4.98E+00
1-ACW-TNK-RP-T4_001__	ACCW SURGE TANK 1-1217-T4-001 RUPTURES CAUSINGLOW LOW LEVEL	3.35E+00

E.3.2 Single Component Failures to Consider as MUCCFs from Level 1 Internal Floods and Level 2 PRA for Internal Events and Internal Floods

There were no potential MU CCFs identified in the Level 1 PRA for internal floods that were not already identified in the Level 1 PRA for internal events.

Several of the significant Level 2 PRA BEs are failures of post-core-damage mitigation strategies represented by human failure events (HFEs). The Level 2 PRA modeling approach uses single HFEs to model post-core-damage mitigation strategies that require both human actions and equipment. Equipment failures are not expected to significantly contribute to the strategy failures and not modeled for the single unit accident sequences. Equipment failures may be significant for dual-unit sequences if both units have demands for the same equipment and resources. The Level 2 PRA report for internal events and floods (NRC, 2022b) describes the modeled actions.

Table E-13 summarizes the results for the identification of new single component failures from the Level 1 PRA for internal floods and the Level 2 PRA for internal events and internal floods that are candidates for cross-unit CCFs. As for Table E-1, the results shown in Table E-13 are ranked by risk importance with the most risk significant components and associated failure modes shown in bold text. Table E-13 is based on Table E-14 (single component failures with FV importance of 0.005 or greater) and Table E-15 (single component failures with RAW of 2 or greater). The only single component failure that is recommended to be considered is the containment flooding EDMG strategy that uses B.5.b pumps and firewater storage tanks. Components with modeled CCFs for the other EDMG strategies (e.g., steam generator atmospheric relief valves (ARVs)) were already identified in Table E-6 for the internal fire Level 1 PRA.

Two significant Level 2 PRA BEs are excluded from the multi-unit CCF consideration because potential for cross-unit CCF failure mechanisms is deemed to be not applicable:

- 1-L2-BE-PZRVSTUCK-SRV: Pressurizer SRVs (1 of 3) fails to reclose after opening on demand. The failure model is based on pressure demands to the relief valves. Each unit’s SRVs are expected to fail independently based on the demands at that unit.

- 1-L2TEAR: Containment liner small leakage due to pre-existing failures during initial design and construction or long-term degradation or lack of preventative maintenance. The failure probability is based on industry analysis in WCAP-15691-NP, Revision 5. It is expected that this failure mode would impact each unit independently.

E.3.3 Single Component Failures to Consider as MUCCFs from Level 1 PRA for Internal Fires, Seismic Events, and Wind-Related Events

There are no additional single components in the single unit model for fires, wind-related events, or seismic events to consider for cross-unit CCFs.

E.3.4 Single Component Failures to Consider as MUCCFs from Level 1 PRA for Internal Events – FLEX Sensitivity Case

Section E.2.4 addressed such potential CCFs. Section E.2.4 discusses how some FLEX equipment could be modeled as both a CCF BE for a single unit and a cross-unit CCF for both units. However, the simplified modeling of the FLEX sensitivity case did not explicitly model any CCFs.

Table E-13 Level 1 PRA for Internal Floods and Level 2 PRA for Internal Events and Internal Floods – Single Component Failures for Multi-Unit CCF Consideration

System	Components (*ordered by risk importance*)
Firewater	Equipment supporting SCG-1 containment flooding: **B.5.b diesel-driven pump** (1 of 2 for the site required), **firewater storage tank** (2 of 2 for the site required), demineralized water storage tank (DWST) (alternate water supply).
Main Steam System	Equipment supporting SAMG strategy SAG-1 to inject to SGs: ARVs (2 of 4 per unit required), Equipment supporting SAMG strategy SAG-2 depressurize RCS: ARVs (assume 4 of 4 per unit required)
Condensate System	Equipment supporting SAMG strategy SAG-1 to inject to SGs: Condensate pumps (assuming 1 of 3 per unit required)

Table E-14 Single Component Failures in Level 2 PRA – FV ≥ 0.005

Level 2 Basic Event	Description	FV
1-L2-BE-MANUALTDAFW-GEN	Failure of Manual Extension of TD-AFW in SBO	4.07E-01
1-L2-OP-SCG1-1	Operator Fails to Carry Out SCG-1 (Spray Containment w/ Firewater)	1.48E-01
1-L2-OP-SAG1	Operator Fails to Carry Out SAG-1 (Open 2 ARVs and Feed SGs)	7.79E-02
1-L2-BE-PZRVSTUCK-SRV	Pressurizer SRVs Do Not Fail Open During CD	2.60E-02
1-L2-OP-SAG2-1	Operator Fails to Carry Out SAG-2 (Open all ARVs - Not Depress)	9.02E-03

Table E-15 Single Component Failures in Level 2 PRA – RAW ≥ 2

CCF Basic Event	Description	RAW
1-L2TEAR	CONTAIN ISOL FAIL DUE TO PRE-EXISTING MAINT ERRORS	2.38E+00

E.4 Results for SFPs and Reactors: CCF Group Expansion

The ISR project team performed a preliminary identification of identical components between the SFPs and the two reactors. This preliminary identification did not find any common components with the two reactors.

In the SFP combined Level 1 and Level 2 PRA, two EDMG mitigation strategies for accident response are modeled, one “internal” and the other “external.” For the internal EDMG strategy, NSCW water and fire protection standpipes and associated valves are needed. For the external EDMG strategy, portable equipment (e.g., B.5.b pumps) is used (which was addressed under the category of “shared or connected SSCs”). Consequently, there were very few types of equipment to review for the category of “identical components.” Since the NSCW and fire protection piping valves are different in design and function than those used in safety-related reactor systems, no “identical components” were identified between the SFPs and the reactors.

Due to project scope limitations, the FLEX case for SFPs was not performed. However, Table E-9 was used to identify the following FLEX equipment that could have been modeled with CCF BEs:

- SFP FLEX submersible pump hydraulic units
- SFP FLEX pump submersible pumps
- sets of monitor spray nozzles for SFP spray and connection equipment

E.5 Summary of CCF Results

The results of this section are used as inputs to the development of the MU and multi-source risk calculations with respect to sitewide dependencies. In particular, the choices on how and what to model regarding cross-unit and sitewide CCFs.

E.5.1 Summary for CCF Group Expansion

Different types of results are shown in Section E.2 for potential cross-unit CCF group expansion for the two reactors. Tables in that section showed results for internal events, internal floods, internal fires, wind-related events, and seismic events.

Table E-1, Table E-2, and Table E-3 provide the results for the Level 1 PRA for internal events. Table E-1 summarizes the systems, components, and failure modes involved in modeled CCFs. This table shows, along with the background details provided in Table E-2 and Table E-3, that there are many potential CCFs that could be expanded to being modeled as cross-unit CCFs. However, bold font is used in Table E-1 to indicate which components and failure modes are most risk-significant, drastically reducing the large number of potential cross-unit CCFs.

A similar approach was taken for the Level 1 PRA for internal floods and the Level 2 PRA for internal events and internal floods, as shown in Table E-4 and Table E-5. However, Table E-4 shows that only one system and associated component (and failure mode) was considered risk-significant (i.e., containment isolation valves fail to operate).

In contrast, tables for potential cross-unit CCFs from the Level 1 PRAs for internal fires (Table E-6), wind-related events (Table E-7), and seismic events (Table E-8) include all possible systems, components, and failure modes, although the results are sorted by FV importance.

Despite the differences in how results are presented, a summary of potential CCF group expansion can be developed based on risk significance. Consideration of risk significance of CCF groups may be important because the number of potential cross-unit CCFs is likely too large for all to be included in the MU risk model or performing MU risk calculations.

Based on the results given in Section E.2, there appears to be some overlap of CCFs for the same systems, components, and failure modes between different PRAs and hazards. This information was considered when deciding on the approach for modeling cross-unit CCFs in the L3PRA project’s MU and multi-source risk calculations.

E.5.2 Summary for Single Components Modeled as MUCCFs

As for the results discussed in the previous section, risk importance measures were used to identify new potential, cross-unit CCFs. Most of the new potential, cross-unit CCFs identified in Section E.3 were identified from the Level 1 PRA for internal events, and none from the Level 1 PRAs for internal fires, wind-related events, and seismic events.

Based on the results given in Section E.3, the following single components modeled in the reactor PRAs were recommended to be modeled as cross-unit CCFs in the MU risk model:

- AFW – turbine-driven pump fails to run
- Electrical – DC bus failure
- ECCS – NCP fails to run
- B.5.b pumps (Level 2 PRA EDMG strategy)
- Firewater storage tanks (Level 2 PRA EDMG strategy)
- 480 V FLEX DGs
- SG FLEX pumps
- Boron injection FLEX pumps
- RCS makeup FLEX pumps
- FLEX fuel tankers
- FLEX tow vehicles
- Makeup FLEX pumps

As noted in Section E.4, the SFPs also require B.5.b pumps (and EDMG strategies) for the base case analysis. Though not modeled for the L3PRA project ISR task, when FLEX strategies are considered, the SFPs also use FLEX pumps.

APPENDIX F
IDENTIFICATION OF SITEWIDE HUMAN AND ORGANIZATIONAL DEPENDENCIES

This appendix presents the results for the Phase 3 sitewide dependency assessment for the category of potential human and organizational dependencies. This assessment was performed as part of the Phase 3 assessment of potential sitewide dependencies that supports the integrated site risk (ISR) task.

F.1 Definition of Sitewide Human and Organizational Dependencies

As stated in Section B.7.3, this category of sitewide dependency has been defined differently in similar guidance for the assessment of potential human and organizational dependencies. The definition used for the L3PRA project's ISR approach is intended to capture all dependencies related to human and organization resources, including potential dependencies that were identified in other categories (e.g., shared physical resources and proximity dependencies) but are more appropriately addressed by human reliability analysis (HRA).

As indicated in Table B-1 regarding the overall approach of assessing potential sitewide dependencies, the definition for the "Human or Organizational" category of dependencies is:

> Dependencies between operator actions across multiple radiological sources that can result from multiple causes, including sharing of staff and shared organizational factors.

The following are examples of potential human and/or organizational dependencies discussed in the EPRI and IAEA reports (EPRI, 2021a; IAEA, 2019, 2021a) that address how to perform multi-unit PRA (MUPRA):

- shared human resources between units
- shared control rooms
- common procedures (e.g., emergency operating procedures [EOPs], abnormal operating procedures [AOPs], severe accident mitigation guidelines [SAMGs], and FLEX procedures[67])
- common operator training
- common human machine interface
- common command and control structure (C&C)
- common technical support center (TSC)
- common emergency response organization (ERO)

[67] 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.

- common offsite support
- increased stress due to multi-unit (MU) accident conditions
- accessibility concerns due to the other unit's degraded condition
- common environmental concerns for operators of both units (e.g., field operators taking actions at local control stations, locations shared by both units, or outside the plant[s])

In addition, typical HRA concerns are relevant, such as:

- timing of the action (especially with respect to when conditions from one reactor can affect another reactor)
- cues and indications to prompt and/or support operator actions
- potential dependencies with prior actions

As a reminder, Section B.7 also states that all potential dependencies identified in the Phase 3 assessment are:

- typically modeled by adjustments to basic event probabilities, rather than logic modeling
- difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether CCF groups should be expanded and what adjustment factor to use for an expanded group)
- difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the TSC, etc.
- typically require modeling that is beyond the PRA state-of-the-art

F.2 Approach for Identifying Potential Sitewide Human and Organizational Dependencies

Section B.7.3 also describes the approach used to identify potential sitewide human and organizational dependencies. Based on the approach described in Section B.7.3, the results for this sitewide assessment of potential human and organizational dependencies are organized by radiological source and by information source The information sources used for the human and organizational sitewide dependency assessment are:

- Phase 1 sitewide dependency assessment results given in Appendix C
- Phase 2 sitewide dependency assessment results given in Appendix D
- Assessment of potential cross-unit common cause failures (CCFs) as part of Phase 3 given in Appendix E

- Information supporting HRAs for the various single radiological source PRAs
- Various single radiological source PRA results

Section B.7.3 discusses two different types of human and organizational dependencies: "explicit" and "implicit." In addition, Section B.7.3.1 states that "explicit" human and organizational dependencies are most likely to be identified from the Phase 2 sitewide dependency assessment and the modeling in the single radiological source PRAs. As stated in Section 6.5 of EPRI (2021a), "… explicit dependencies [associated with, for example, shared SSCs] are judged to dominate over these implicit dependences [such as] shared accident sequences." The L3PRA's ISR task also focuses on addressing "explicit" human and organizational dependencies.

It should be noted that the identification of a potential human or organizational dependency does not automatically require that such a potential dependency be modeled. Limitations, such as the state-of-the-art, may preclude such modeling. Rather, this identification should be viewed similar to that required by the PRA standard (see, for example, ASME/ANS [2022]) for identifying potential sources of uncertainty.

Section F.3 presents the results of the sitewide assessment of human and organizational dependencies for the two reactors. Section F.4 addresses potential sitewide dependencies between spent fuel pools (SFPs) or dry cask storage (DCS) with the two reactors.

F.3 Results for Two Reactors: Potential Sitewide Human and Organizational Dependencies

Results from the assessment of potential sitewide human and organizational dependencies is given below when considering only the two reactors on site. As stated above, the results are provided in different sections for the different information sources used in the sitewide human and organizational dependency assessment.

F.3.1 Results for Potential Sitewide Human and Organizational Dependencies from Phase 1 Sitewide Dependency Assessment

Results for the Phase 1 sitewide dependency assessment concerning multi-unit IEs (MUIEs) or sitewide IEs are given in Appendix C. These results show that almost all the MUIEs and sitewide IEs occur due to offsite causes. Human and organizational dependencies could be the cause of other MUIEs or sitewide IEs. Human-induced IEs cannot be easily categorized as either "explicit" or "implicit" human and organizational dependencies.

In general, HRA is typically not concerned with human-induced IEs unless the operator response to the human-induced IE is more difficult than the parallel IE with hardware or external causes. Consequently, human-induced IEs are seldom considered explicitly in PRA[68] except for low-power and shutdown conditions (e.g., draindown events). The scope of the L3PRA project's ISR task does not include low power and shutdown.

However, there is one potential sitewide IE identified in the Phase 1 sitewide dependency results (see Table C-7 in Appendix C) that could merit attention for potential human and

[68] Human-induced IEs are captured with other causes of IEs in IE frequency data.

organizational dependencies: loss of nuclear service cooling water (NSCW). For the L3PRA project, the frequency of this IE was determined using fault tree (FT) modeling, especially considering different combinations of CCFs for the NSCW system. In addition, since only two of six NSCW pumps are normally running, operator action to start additional pumps is included in the FT for this IE. In principle, common factors could result in failed operator actions for the NSCW system that affect both reactor units (and, for some cases, the SFPs). It could be argued that this modeling is an "explicit" human and organizational dependency.

Section F.3.6 below discusses a few operational events for loss of cooling that provide insights into how such operator failures can occur. This discussion indicates that the underlying causes of these events typically include very specific pre-accident conditions and plant-specific factors (e.g., organizational factors). Because of these underlying causes, it could be argued that these represent "implicit" human and organizational dependencies. Since the current scope of the ISR task does not allow for such investigations for the reference plant, no further investigation is recommended for this potential human and organizational dependency.

F.3.2 Results for Potential Sitewide Human and Organizational Dependencies from Phase 2 Sitewide Dependency Assessments

Appendix D provides the results for the Phase 2 sitewide dependency assessment. These results identify potential sitewide dependencies for the two reactors, SFPs, and DCS in two categories:

- shared physical resources (discussed in Section F.3.2.1)
- shared or connected structures, systems, structures and components (SSCs) (discussed in Section F.3.2.2)

In most cases, there are explicitly modeled human failure events (HFEs) in the single radiological source PRAs associated with the physical resources and SSCs identified in Appendix D. Since physical resources and SSCs that are shared or connected between the reactors might be modeled as cross-unit HFEs in a MU risk model, such dependencies could be termed as "explicit" human and organizational dependencies. However, underlying causes for the human and organizational dependencies may include causes that would match the "implicit" factors discussed in Section B.7.3.3.

F.3.2.1 Results From Identified Shared Physical Resources

Table D-1 in Appendix D for the Phase 2 sitewide dependency assessment documents the shared physical resources between the two reactors for all hazards and PRA types. Three different shared physical resources are identified in this table:

- 500 kV and 250 k V switchyards
- alternate switchyard
- fire water storage tanks (FWSTs) (north and south)

There are different implications for each of these physical resources with respect to HFE modeling and cross-unit dependencies:

- Switchyards: For example, the same operator actions taken to restore switchyard-related losses of offsite power (LOOPs) for one reactor also restores power for the second reactor. ***Such operator actions should be modeled as single actions that affect both units.***

- Alternate switchyard: Only one reactor can be connected to the alternate switchyard. So, for certain LOOPs, the second reactor will not have offsite power while the first reactor will.

- FWSTs: The FWSTs are called out for use when implementing the extensive damage mitigation guidelines (EDMGs), as noted in the Level 2 PRA report for internal events and floods (NRC, 2022b). However, the Level 2 PRA defines "success" as the use of both FWSTs for a single reactor. Consequently, the FWSTs can be used for only one of the two reactors and the operator action (and associated EDMG strategy) is no longer feasible for the second reactor if the FWSTs have been used for the first reactor.[69] For example, according to the Phase 2 sitewide dependency assessment results documented in Appendix D, "*the needed volume of water for success of such EDMG strategies is assumed to be equivalent to both FWSTs. However, the smaller volume DWST [demineralized water storage tank] is indicated to be an option, too. It is not currently known whether EDMG strategies can be successful with the smaller volume DWST.*"

In summary, in the case of the switchyards, HFEs modeled to represent power restoration should be modeled as common to both Units 1 and 2. For the alternate switchyard, only one unit can credit it as a power source. For the FWSTs, per the success criteria, it is not feasible for both reactors to be adequately fed by these water tanks.

F.3.2.2 Results From Identified Shared or Connected SSCs

Table D-3 documents the results for the Phase 2 sitewide dependency assessment for shared or connected SSCs for the two reactors, for all hazards and all PRA types. Only one type of component[70] and eight structures are identified in this table.

The single type of shared component is the B.5.b pump and associated equipment. Table D-3 provides the following important information on these components:

- There are two B.5.b pumps (and associated equipment) to implement EDMG strategies. However, one B.5.b pump is stored nearby (in the warehouse) while the other is at the fire training facility (farther away).

- While, in principle, two B.5.b pumps for two reactors should be sufficient, it is not known if there is adequate time and other resources to use the second B.5.b pump that is located farther away from the reactors and associated connection points.

Consequently, there are questions about the feasibility of both reactors being fed by the B5.b pumps due to potentially inadequate staffing, potentially unavailable equipment to support B.5.b

[69] There is mention of refilling the FWSTs but there are no specifics on how this is done.

[70] The emergency diesel generators are also identified in Table D-3. However, in the notes, it is recommended that these components not be included in the ISR task since the only way they can be "shared" is if they are cross-tied and such cross-tying is not modeled in the PRAs.

operation, and potentially inadequate time to transport the second B.5.b pump to where it is needed for EDMG strategy implementation.

Some key points are provided below for the eight structures according to their characteristics and likely treatment for dependencies. In all cases, dependencies related to seismic events are potentially important and are addressed in Section F.3.3.4 under the topic of hazards correlations. Also, some of the shared or connected structures may have important dependencies for certain fire events.[71] However, none of the shared or connected structures represent important "explicit" human and organizational dependencies. For some of these structures (e.g., shared main control room), potential "implicit" human and organizational dependencies are discussed in Section F.3.4.

- The auxiliary and turbine buildings for both units are connected. However, as indicated in the notes provided in Table D-3 in Appendix D, these buildings are separated in such a way that dependencies are not likely to be important except for seismic events.

- The FLEX building is common to both units. However, as noted in Table D-3 of Appendix D, "FLEX buildings are designed and constructed to withstand external hazards. Consequently, it is unlikely that they would fail for these or any of the internal hazards."

- The main control rooms are essentially a shared space for both units. "Implicit" human and organizational dependencies regarding this sharing are discussed in Section F.3.4. For fire HRA/PRA, this sharing is important; for example, in the case of control room fires (i.e., it is assumed that both control rooms must be abandoned due to uninhabitable conditions). Also, this sharing could be important for seismic events.

- As stated in Table D-3 in Appendix D, "Unit 1 and Unit 2 share the control building although there is some separation by walls and doors…Because of [this] separation …, the sharing of the control building for Unit 1 and Unit 2 is not expected to be an important dependency for internal events….The control building connections may be relevant for MU fire PRA…[these] connections are likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations)."

- The TSC is common to both units, is located in the control building, and has the same dependency assessment as for the control building.

- As stated in Table D-3 in Appendix D, "[t]he fuel handling building is common to Units 1 and 2. The fuel handling building houses both units' spent fuel pools, which are normally connected through the cask loading pit. Because the SFPs are considered a separate radiological source in calculating multi-source risk, the fuel handling building is not considered a shared structure for this analysis. [These connections are] likely relevant for seismic PRA but will be treated under Phase 3 sitewide dependencies (e.g., hazard correlations)."[72]

[71] The identification and treatment of these dependencies is specific to the multi-unit fire PRA modeling. Discussion of fire-specific human and organizational dependencies is given in the documentation of the multi-unit fire risk calculations.

[72] When the SFPs are considered, discussion of potential impacts on operator actions will be addressed.

- As stated in Table D-3 in Appendix D, “[t]here are two cable spreading rooms for each unit (four rooms in all). The MCRs are on Level 1...[t]he rooms are separated between units with a door between them...[b]ecause of [this] separation…the connections between the cable spreading rooms for Unit 1 and Unit 2 are not expected to be an important dependency for internal events and internal floods. Per information supporting the utility’s fire PRA, smoke from fires in the cable spreading rooms will not cause MCR abandonment. However, like the connected auxiliary and control buildings, the connections between the cable spreading rooms may be relevant for MU fire PRA” and is expected to be important for seismic PRA.

F.3.3 Results for Potential Sitewide Human and Organizational Dependencies from Other Phase 3 Sitewide Dependency Assessments

The Phase 3 assessment of sitewide dependencies addressed the following categories of potential sitewide dependencies:

- potential CCF group expansion (e.g., cross-unit CCFs)
- proximate dependencies
- cascading failures (i.e., failures that propagate from Unit 1 to Unit 2 due to dependencies)
- human and organizational dependencies
- potential hazards correlations

The identification of potential sitewide human and organizational dependencies (i.e., the fourth type of Phase 3 dependencies) considered the results from each of the other types of Phase 3 potential sitewide dependencies, as discussed in the sections below.

F.3.3.1 Potential CCF Group Expansion

There is no state-of-practice modeling of a connection between CCFs and post-accident operator actions. Any human component to the cause of cross-unit CCFs (e.g., a human-caused CCF due to improper maintenance) would already be represented in their modeling. Operator responses to cross-unit CCFs should not be different than that of single component failures. Consequently, there are no human and organizational dependencies associated with CCFs that are recommended to be modeled.

F.3.3.2 Proximity Dependencies

From Table B-1, proximity dependencies are defined as “[d]ependencies that arise across radiological sources from: (1) exposure of multiple SSCs to shared phenomenological or environmental conditions, (2) common features between units, or (3) operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source.” Except for those dependencies associated with operator actions, this category of dependency is addressed in Appendix G. Proximity dependencies that impact human actions may change the feasibility of an operator action (i.e., represent an “explicit” dependency) or might be represented as a less severe influencing factor in HFE quantification (i.e., an “implicit” dependency).

Table B-1 further provides the following information under the headings of “example(s)” and “how modeled”:

- *Example(s):* Failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat or cold, radiation levels), debris, explosions, etc., from a nearby radiological source.

- *How Modeled:* Environmental conditions that impact operator actions can be modeled similarly to SSCs but should be addressed as “human or organizational dependencies. The timing of the conditions from one reactor that can affect another reactor is also important. Once such dependencies are identified as impacting SSCs or operator actions, one approach would be to assign conditional failure probabilities to basic events or HFEs (e.g., the basic events and HFEs in Unit 2’s risk model can be altered due to failures, environmental conditions, debris, explosions, etc., that exist for nearby Unit 1).

Because the plant areas for the two reactor units are mostly separated (except for the control building and turbine building), operator actions taken outside the MCR in the Level 1 PRAs for internal events and internal floods are not expected to have any dependencies. This is relevant also in cases when one reactor experiences core damage in advance of the other reactor (i.e., the action locations related to the second reactor should not be in proximity of the affected areas for the first reactor). Also, any environmental conditions associated with external hazards should be addressed through the consideration of hazards correlations (see Section F.3.3.4).

The existing fire PRA has identified scenarios that require both MCRs to be abandoned due to environmental conditions. Specifically, if a fire affects the habitability of either MCR, both MCRs are treated as being affected since the MCRs are connected. In addition, the existing fire PRA has identified scenarios in which a fire can cascade from one unit to the other (see Section F.3.3.3). When MU risk is calculated, the analyst needs to be certain that all credited operator actions remain feasible (i.e., no actions are required in or near the fire location).

For Level 2 PRAs, high radiation levels from a reactor post-core-damage are possible in some locations. To support Level 2 HRA, a habitability assessment was performed to identify areas inside the plant that could experience high radiation levels. However, no information on radiation levels outside the plant was available to the project team. Some plant personnel interviewed during the 2014 visit to the reference plant site raised this as a possible concern for the single unit PRA. It is possible that, if such radiation levels existed, operator actions to implement EDMG strategies for both units could be affected. The operator actions could be delayed (e.g., waiting for health physics personnel to perform radiation surveys) or could be rendered infeasible (i.e., radiation levels too high to attempt the action).

F.3.3.3 Cascading Failures

The results for Phase 2 sitewide dependency assessment given in Appendix D already indirectly addressed cascading failures. In particular, Section D.4 states that “the two reactors are considered to be “loosely coupled” for all hazards except certain fire scenarios and seismic events,” per the definition provided in the EPRI (2021a). Appendix G also discusses the applicability of cascading failures for the reference site. Per the guidance in EPRI (2021a), cascading failures likely do not need to be addressed for “loosely coupled” reactors. Consequently, except for fires and seismic events, the ISR task does not address potential cascading failures.

For fires, there are sequences in the existing single unit fire PRA that include a fire propagating to the other unit. However, operator actions in a fire PRA can only be credited in locations without fire or smoke. Consequently, it is not expected that there are direct or explicit dependencies between Unit 1 and Unit 2 operator actions even for these scenarios.

For seismic events, the “hazards correlations” category of Phase 3 dependency assessments addresses this concern.

F.3.3.4 Potential Hazards Correlations

The influence of hazards on operator actions has already been addressed in the existing single unit PRAs for floods, fires, and external hazards. Changes in human error probabilities (HEPs) to account for hazards are typically small unless the hazard affects the feasibility of the operator action. For example, relatively significant multipliers are often used for HEPs associated with operator actions taken in the higher seismic bin scenarios when significant structural damage is expected on site (see Table 5.3-3 in the L3PRA project’s Level 1 seismic PRA [NRC, 2023b]). If the influence of a hazard encompasses both units, especially in the same way, it is recommended that the operator actions continue to be treated as independent. For the lower seismic bins for which little sitewide damage may have occurred, operator actions should be similar to that for internal events. For the higher seismic bins, HEPs are already high and frequency inflation is a problem for both CDF and MUCDF calculations. Consequently, there would not be any benefit to re-assessing already high value HEPs.

F.3.4 Results for Potential Sitewide Human and Organizational Dependencies Identified from HRA Information

The best way to evaluate the potential “indirect” or “implicit” human and organizational dependencies is to draw upon the various information collected and interpreted for the existing HRAs performed for the various single unit PRAs. Table F-1 documents this evaluation based upon the HRAs, plant site visits, and other HRA-relevant information. Modeling these dependencies is beyond the current state-of-practice, but identifying these possibilities is considered good practice. The treatment of potential “indirect” human and organizational dependencies is a candidate for future research.

As shown in Table F-1, there are both positive and negative impacts that are possible for most of these potential dependencies. However, the recommended assessment is generally that commonalities or dependencies should be considered to have a positive effect. These results are consistent with discussion in Section 6.5 of EPRI’s MUPRA report (EPRI, 2021a).

When the MU risk model is quantified, it is possible that some of these potential dependencies could be explored in sensitivity studies. One area that might be worth exploring is the Level 2 PRA operator actions associated with SAMGs and EDMG strategies. The Level 2 HRA, while justifying that such operator actions should be credited, did identify that these procedures provide less support to operators and are trained on less frequently than other procedures used by operators (e.g., EOPs). Also, unlike the FLEX[73] procedures, SAMGs and EDMGs were not developed for a sitewide response involving all radiological sources simultaneously.

[73] FLEX refers to the U.S. nuclear power industry's proposed safety strategy, called Diverse and Flexible Mitigation Capability. 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 offsite.

Table F-1 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies

Characteristic of Potential Dependency	Does the Dependency Exist at the Reference Plant? (Yes/No)	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Shared MCR	No. The MCRs are connected physically by essentially an "open door," but they are separated by a relatively large distance with respect to control locations. If shift supervisors from each unit wanted to share information, it would only require a short walk.	If MCRs were shared, operators could be distracted by alarms on the other unit; "group think" that is incorrect; loss of "swing" operator. But, given the MCRs are not shared and there is considerable separation of control boards and operators for the reference plant, such distraction is very unlikely.	Because travel between from the Unit 1 and Unit 2 MCRs is quick and easy, the following is possible: face-to-face communication, "group think" that is correct, sharing a "swing" operator, and closer coordination between units.	EPRI (2021a) did an initial comparison between shared and connected MCRs and preliminarily did not find any significant differences between the two.
Connected MCR	Yes	See above	See above	See above
Common procedures	Yes. The essentially identical units have essentially identical EOPs, SAMGs, EDMGs, FLEX procedures, fire response procedures, maintenance procedures, etc.	If there is a weakness, it likely will affect actions for both units.	If the procedural support is good, actions should be independent.	Weaknesses or "gaps" might be considered for explicit modeling (e.g., if action for one unit is failed due to such a "gap," then the same action for second unit probably should be considered "failed" also).
Common training	Yes	Same as for "common procedures"	Same as for "common procedures"	Same as for "common procedures"
Common human-machine interface	Yes	Same as for "common procedures"	Same as for "common procedures"	Same as for "common procedures"
Common C&C	Yes and No. Each unit has its own shift supervisor. There is one unit supervisor for both units. See "common TSC" for C&C assessment when emergency director (ED) responsibilities shift.	Same as "connected MCR"; challenge of responding to multiple reactors within the same time period. However, eventually C&C shifts responsibility to a single ED for both units.	Similar to "connected MCR," "common procedures," and "common training"	EPRI (2021a) suggests that on-site C&C should be a net positive.

Table F-1 Assessment of Implicit/Indirect Potential Human and Organizational Dependencies (cont.)

Characteristic of Potential Dependency	Does the Dependency Exist at the Reference Plant? (Yes/No)	Potential Negative Impacts	Potential Positive Impacts	Notes for Potential Modeling
Common TSC	Yes	By the time the TSC is staffed, the responsibility of ED should be shifted to someone located in the TSC. From 2014 interviews of managers who could take the ED role after transfer into SAMGs,[74] it is expected that the TSC will be staffed with twice as many personnel if the site is responding to a dual-unit event. In addition, the HRA team learned that all four managers who could take the ED were licensed SROs, or had been licensed SROs, at the reference plant.	Similar to “connected MCR,” “common procedures,” “common training,” and “common C&C” In addition, the HRA team learned during the 2014 plant site visit, that many of those who have responsibilities in the TSC have worked at the reference plant for their whole careers and, therefore, have a strong understanding of the reference plant and its operating history.	Same as for “common C&C”
Common ERO	Yes	Same as for “common C&C” and “common TSC”	Same as for “common C&C” and “common TSC”	Same as for “common C&C” and “common TSC”
Common offsite support	Yes	No information was collected for the L3PRA project on the offsite organization.	No information was collected for the L3PRA project on the offsite organization.	
Increased stress due to MU accident	Likely	No specific information regarding stress in MU accidents was collected for the L3PRA project.	No specific information regarding stress in MU accidents was collected for the L3PRA project.	Depending on the severity of the MU event, increased stress could be a reasonable assumption. However, given the implementation of FLEX strategies, the additional training and attention may offset the potential stress for some severe accidents.

[74] The 2014 plant site visit for HRA included discussions of potential sitewide events even though the primary purpose was to support Level 2 HRA for internal events PRA.

F.3.5 Results for Potential Sitewide Human and Organizational Dependencies Identified from Single Radiological Source PRA Results

Each PRA type and hazard is discussed briefly below. The focus of this discussion is primarily on “explicit” human and organizational dependencies.

For most of the Level 1 PRAs, cross-unit human and organizational dependencies should not be a concern. This assessment is consistent with guidance given in Section 6 of the EPRI MUPRA report (EPRI, 2021a). The MCRs are connected, but working areas are separated (see Section F.3.4). Operator action locations outside the MCR are even more separated. In summary:

- No cross-unit human and organizational dependencies are expected for internal events and internal flooding Level 1 PRAs.

- Any dependencies affecting operator actions for external hazards are likely to be related to environmental conditions created by the hazard, which can be considered independently.

The situation is different for the Level 1 PRA for internal fires. As stated in Section F.3.4, the MCRs are connected such that smoke from a fire affecting one unit affects the other unit. Consequently, if MCR abandonment is required for either unit, both units must abandon the MCR. Also, Appendix E for the Phase 2 sitewide dependency assessment (e.g., note “d” in Table D-3 regarding the results for shared or connected SSCs and Table D-5 for results of “coupling”) identifies the potential for certain fire scenarios to cascade from one unit to the other.

There are only a few operator actions considered in the Level 2 PRAs, and they mostly involve implementation of SAMGs and EDMGs. The operator actions taken inside plant buildings can be considered to be independent, as for the single radiological source Level 1 PRAs. Independence is less certain for actions taken outside plant buildings. In summary:

- As documented in Appendix D, limited physical resources (both availability of field operators and B.5.b pumps) may result in only one of the two reactors being able to implement EDMG strategies.

- As noted in Section F.3.3.2 above for proximate causes, it is possible that radiation levels from one reactor already experiencing core damage may prevent operator actions for the other unit.

“Explicit” human and organizational dependencies between the reactors for the FLEX sensitivity cases are not expected. The design of these strategies, including detailed timelines for every operator, piece of critical equipment, and activity, supports treatment of each operator action as being independent.

There are no operator actions to consider for the Level 3 PRAs (i.e., the consequence analyses).

F.3.6 Results for Potential Sitewide Human and Organizational Dependencies Identified from Operating Experience

Most research efforts related to HRA include a review of operational experience to assist in identifying potentially relevant issues for operator performance. For example, the 2019 IAEA report on MUPRA (IAEA, 2019) includes a review of events such as the 2011 Great Japan Earthquake and associated Fukushima accident as well as the Blayais flooding event. However, it could be argued that current FLEX strategies have, in principle, addressed the issues identified in such events.

The question is whether there are still relevant HRA issues to be identified in historical experience, including events that were not sitewide events. As such, further investigation of relevant historical events is recommended as a candidate for future research.

F.4 Results for SFPs and DCS: Potential Sitewide Human and Organizational Dependencies

There are no identified sitewide human and organizational dependencies between the DCS facility and the two reactors. There are no shared resources, shared or connected SSCs, operator actions, shared procedures, and so on.

For the SFPs and the two reactors, the identification of shared resources and shared or connected SSCs given in Appendix D, as well as the identification of potential sitewide CCFs given earlier in this appendix, show that there is an intersection between the SFPs and the reactors in the EDMG and FLEX strategies. In particular, accident mitigation for the reactors (in the Level 2 PRA) and for the SFPs requires, in both cases, implementation of the EDMG strategies. In turn, the EDMG strategies in both cases require operator actions (and associated equipment). Similarly, implementation of the FLEX strategies requires operator actions and associated equipment.

Particularly, in the case of EDMG strategies, it is not clear that planning has been done to be certain that there is adequate staffing (as well as equipment and water resources) to address the needs of both the SFPs and the reactors. In contrast, most FLEX implementation plans include a detailed integrated timeline that shows all staff and equipment that must be addressed throughout the accident sequence. In addition, the reference plant Final Integrated Plan provides an extensive list of water resources to address the needs of all radiological sources on site.

F.5 Modeling Implications of Results for Potential Sitewide Human and Organizational Dependencies

Based on the results of the sitewide assessment of human and organizational dependencies, it is recommended that such dependencies for the L3PRA project be treated consistently with recent previous guidance on MUPRA.

In particular, Section 2.3.5 of EPRI (2021a) states the following:

- “Similar to CCF aspects discussed previously, HRA can be a significant driver due to subjectivity in deriving both the qualitative and quantitative factors associated with human error probabilities for MU issues (as well as lack of data). In addition, actions associated with mitigating core damage in one unit while other units on-site are proceeding toward severe accident scenarios (core damage progression and potential releases, as observed at Fukushima Daiichi) can be particularly challenging. This may be an aspect which is currently beyond the state-of-art of PRA methods and may need to be addressed qualitatively (for example, by understanding how MU aspects are addressed in procedures and training, rather than attempting full quantification). Some HRA aspects are already challenging for SU PRA issues (for example, main control room abandonment scenarios, manual operator actions to be performed during severe weather conditions), and they are bound to be compounded for MUPRA purposes.

Similar to CCF issues, if MUPRA modeling identifies specific operator actions as drivers for MU risk, it is worth considering if (1) there is sufficient confidence in the underlying bases for the results for risk-informed decision-making (RIDM) purposes, and (2) if so, that some operational aspects are considered as potentially justified improvements."

Section 6.5.1 of EPRI (2021a) adds the following relevant statements:

- "MU accidents start with a MU initiating event. The most likely (i.e., most probable) condition for each unit is that they are both going down the same accident sequence path. This is due to the modeling of common cause failures of common component-types across units. For example, with a MU LOOP initiator and Unit 1 EDGs failed, it is more likely the Unit 2 EDGs will also fail—in contrast to any other component failures in Unit 2 that would lead to core damage. Thus, the units "share" accident sequences with implicit dependencies due to the increased probability of failure of common component-types across units.

  Thus, if each unit is on the same accident sequence given a MU initiating event, any required operator action would be expected to occur in each unit with about the same plant context (time window, cues, competing demands) and with the same resources (AOPs, EOPs, training, experience). It may be that much of this implicit dependence is driven by modeling assumptions (e.g., CCF parameters for large groups) and lack of knowledge regarding details of accident sequence timing."

However, different choices on modeling and representing human and organizational issues in the L3PRA project's ISR task may be needed when MU or multi-source risk is determined for specific hazards or PRA types. For example, calculating MUCDF may be simplified if certain operator actions are assumed to be completely dependent. On the other hand, the impact of alternatives to the recommendations provided in this appendix could be explored with sensitivity studies in future work.

APPENDIX G
IDENTIFICATION OF OTHER PHASE 3 SITEWIDE DEPENDENCIES

This appendix presents the results for the Phase 3 sitewide dependency assessment that were not addressed in Appendix E and Appendix F. In particular, this appendix addresses the following categories of potential sitewide dependencies:

- proximity dependencies (Section G.2)
- cascading failures (i.e., failures that propagate from Unit 1 to Unit 2 due to dependencies) (Section G.3)
- potential hazards correlations (Section G.4)

Section G.5 identifies some scenarios needing special attention regarding potential Phase 3 sitewide dependencies during later steps in the ISR task.

The assessments described in this appendix were performed as part of the Phase 3 assessment of potential sitewide dependencies that supports the integrated site risk (ISR) task. Generally, only other Phase 3 sitewide dependencies between the two reactors on the reference site are considered here (i.e., dependencies involving the spent fuel pools [SFPs] or dry cask storage [DCS] are not considered).

G.1 Approach

Section B.7 provides guidance on identifying Phase 3 categories of sitewide dependencies. In the approach described in Appendix B, individual analysts who were most knowledgeable of the various L3PRA project models (e.g., different PRA hazards and types) performed the reviews of the respective PRAs and associated materials. Examples of information resources needed for the Phase 3 assessments include the relevant PRA models and associated results (e.g., cutsets), as well as site layout drawings, building layouts and elevations, systems documentation, staffing plans, and procedures.

At present, there is little specific guidance for the Phase 3 categories of potential sitewide dependencies addressed in this appendix. For the L3PRA project's ISR task, this guidance is given in:

- Section B.7.2 for proximity dependencies
- Section B.7.4 for propagation between units
- Section B.7.5 for hazards correlations

As a reminder, Section B.7 also states that potential dependencies identified in the Phase 3 assessment are:

- typically modeled by adjustments to basic event probabilities, rather than logic modeling
- difficult to assess since there is insufficient data upon which to base appropriate modeling (e.g., lack of data to inform whether common-cause failure groups should be expanded and what adjustment factor to use for an expanded group)

- difficult to assess since there is insufficient operational experience upon which to base adjustments to human error probabilities due to common procedures and common training, input from the technical support center, etc.

- typically require modeling that is beyond the PRA state-of-the-art

G.2 Proximity Dependencies

Potential proximity dependencies for the two reactors on the reference site were examined. *Later work will address such potential dependencies with the spent fuel pools (SFPs) (e.g., high radiation fields around SFPs that prevent or make more difficult any operator actions related to the two reactors).* Due to the location of the dry cask storage facility, proximity dependencies between it and the other radiological sources on site are not considered to be likely.

As stated in Section B.7.2, proximity dependencies for the two reactors arise from:

- exposure of multiple SSCs to shared phenomenological or environmental conditions

- common features between units

- operator action locations becoming uninhabitable due to the environmental conditions of a nearby radiological source

Proximity dependencies may cause failure of SSCs and/or operator actions for one radiological source due to SSC failures and/or environmental conditions (e.g., heat or cold, radiation levels), debris, explosions,[75] etc., from a nearby radiological source. External hazards may fail identical or similar structures due to common location of structures for both units. These dependencies are not likely to have been identified in individual risk models for each radiological source.

External hazards and radiological concerns (e.g., conditions associated with the Level 2 PRA) are likely to be the principal concern for SSCs in both units that share phenomenological or environmental conditions. Dependencies related to common features between units (e.g., common location of structures for both units) are likely to be important only to external hazards.

As stated in Appendix B and Appendix F, proximity dependencies that are related to operator actions are addressed in Appendix F. However, like the assessment of other potential sitewide dependencies, dependencies related to operator actions may be initially identified when evaluating a different category of sitewide dependencies, especially if the environmental condition can affect both an SSC and an associated operator action.

Given the discussion above of proximity dependencies, the following contexts that result in SSC failures were searched for:

- common conditions for SSCs for both reactors (e.g., due to the same hazard or response to the same hazard)

- conditions created by one reactor that affects SSCs for the second reactor

[75] The L3PRA project has not produced any results that include the potential for explosions. Section 8 mentions the possibility of hydrogen explosions in the context of Level 2 PRA.

Commonality for the two reactors on the reference site include:

- identical or similar design (e.g., layout and design of the plants, dimensions or sizes of SSCs)
- common or shared locations (such as those identified in the Phase 2 sitewide dependency assessments)
- traditional application of hazard correlations (e.g., modeling identical response for both reactors to the same external hazard)

However, the likelihood of proximity dependencies for the reference site is limited by:

- Separation or independence of most SSCs modeled (i.e., the Phase 2 sitewide dependency assessment indicated that there are few shared or connected SSCs between the two reactors on the reference site)
- Few conditions (e.g., only those caused by fires, internal floods, external hazards, or radiation) can catastrophically affect SSCs in both reactors

A consequence of the above limitations is that proximity dependencies for SSCs due to environmental conditions (that are not associated with external hazards) can only occur for the reactors at the reference site if SSCs for both reactors are shared or connected.

No specific scenarios have been identified at this time that definitively involve proximity dependencies alone. However, characteristics of scenarios, such as those indicated above, will be considered when the multi-unit (MU) model is developed, and associated calculations are performed. As for other PRA models, the results of the MU model will be reviewed to identify such potential dependencies, then be considered for model adjustments.

Some scenarios involving external hazards and potential proximity dependencies are discussed in Section G.4. Additional scenarios that capture some of these characteristics and include elements of proximity dependencies and hazards correlations are given in Section G.5. At this point in the overall ISR task, scenarios that involve radiation as a hazard have not been addressed. Such scenarios are addressed in developing MU Level 2 results.

G.3 Cascading Failures

As for proximity dependencies, the focus of the Phase 3 sitewide dependency assessment for cascading failures was on the two reactors. Because the DCS facility is independent of other radiological sources and is remotely located, it is unlikely that failures could cascade from the DCS to the other radiological sources. However, future work could address potential failures that could cascade from the SFPs to the reactors.[76]

Except for certain fire scenarios that were discussed in Appendix C for the identification of MU initiating events, failures of one unit propagating to another unit is not expected for the reference

[76] During development of the SFP Level 1 and 2 PRAs, scenarios that involve implementation of procedures that use water inventory from the reactors to restore water level in the SFPs were discussed.

site. This expectation is based primarily on the assessment of “loose coupling” between the two reactors given in Appendix D for Phase 2 sitewide dependency assessments.

G.4 Potential Hazards Correlations

Potential dependencies between the two reactors with respect to hazards correlations (and/or proximity dependencies) for external hazards were assessed. Such dependencies related to external events and the SFPs are addressed in the development of sitewide scenarios in a later ISR task, as are dependencies associated with Level 2 PRA.

Note that the examination discussed here focused on information about the site layout, plant design, and so forth that supports the identification of potential dependencies. This assessment considered seismic events (Section G.4.1), wind-related events (Section G.4.2), external flooding (Section G.4.3), internal flooding (Section G.4.4), internal fires (Section G.4.5), weather-related losses of offsite power (LOOPs) (Section G.4.6), and other hazards (Section G.4.7).

G.4.1 Results for Potential Seismic Correlation Between SSCs of Two Reactors

Seismic correlation for SSCs between the two units should be considered and are warranted whenever necessary for MU seismic initiating events. As the intensity of the seismic event increases (i.e., for higher seismic bins), the likelihood of MU seismic correlation increases. Although a seismic correlation model exists for the single unit SSCs and is already included, a two-unit seismic correlation model does not exist. A simple two-unit seismic correlation model that can be introduced by examining the Unit 1 seismic CDF cutsets may be sufficient to capture the impact of this failure mode.

Potential inter-unit seismic correlation is related to both the hazard and the proximity, as noted in Section G.2.

G.4.2 Results for Potential Correlation of Wind-Related Hazards for Two Reactors

Wind-related SSC failures may affect a second unit, given they affected the other unit. During the wind-events walkdown, no major safety system failures due to wind events were identified. The walkdown scope did not include examination of the impact of a wind-related structure failure on another structure belonging to the other unit. At this time, no wind-related MU failures due to proximity are envisioned. (However, there is a scenario involving switchyards listed in Section G.5 for potential scenarios that require “special attention.”)

G.4.3 Results for Potential Correlation of External Flood-Related Hazards for Two Reactors

External flooding was assessed in the L3PRA project for Unit 1 as part of the “other hazards” evaluation and was screened out without detailed modeling. Other than a possible impact on the turbine building shared by both units (proximity), this hazard is not further pursued for MU impact (hazard and proximity-wise) due to its low expected risk impact compared to other MU events.

Also, the plant position to screen external flooding (based on a flooding-focused evaluation) was accepted by an NRC safety evaluation.

G.4.4 Results for Potential Correlation of Internal Flood-Related Hazards for Two Reactors

For the L3PRA project, potential internal flooding for both units was considered in the Phase 2 sitewide dependency assessment. While shared structures and potential flooding scenarios were identified, these scenarios were recommended to be screened out.

G.4.5 Results for Potential Correlation of Internal Fire-Related Hazards for Two Reactors

Internal fires are modeled for Unit 1 only. Although both Unit 1 and Unit 2 fire zones are included for Unit 1 CDF calculations, information to evaluate the potential effect on both units is not available. One useful piece of information available from the L3PRA project's initial Unit 1 multi-compartment fire analysis is that the results support the statement in the reference plant fire PRA that "the [two u]nits [on the reference site] are very well compartmentalized with most boundaries containing fire rated barriers. Therefore, multi-compartment fires have a negligible impact on total plant risk."

However, there is an MCR fire scenario recommended for "special attention" in Section G.5.

G.4.6 Results for Potential Correlation of Internal Events Weather-Related Loss of Offsite Power for Two Reactors

The list of Unit 1 internal initiating events contains a category named "weather-related loss of offsite power" (LOOPWR). This category includes LOOPs caused by weather-related events (ice, snow, salt, lightning, wind,[77] etc.). The initiating event frequency and the offsite power recovery is modeled using actuarial data. This actuarial data can be examined to identify those events that would cause MU LOOPs.

G.4.7 Results for Potential Correlation of Other Hazard Categories for Two Reactors

There are other hazard categories included to some degree of detail in the L3PRA documentation (see NRC [2023a]). These other categories are deemed to have a lesser contribution to MU risk than those categories modeled in detail, and they are not evaluated here. However, it is recognized that, for example, airplane accidents can potentially damage multiple structures belonging to two units at the same time.

G.5 Special Attention Needed for Certain Scenarios

The following scenarios should receive special attention regarding potential Phase 3 sitewide dependencies during later steps of the ISR task:

- *MCR failures with or without MCR abandonment:* Since both units share a common MCR (partitioned only by a minor barrier), an internal fire event impacting MCR for a unit has a strong potential to impact the other MCR, both HRA-wise and equipment-wise. If an MCR evacuation scenario occurs for one unit, a complete correlation between the two units can be assumed. This is an example of a hazard and proximity-related case.

[77] It should be noted that there is a possible overlap between wind events in LOOP-WR and the wind-related events hazard category.

- *Potential two--unit interactions during station blackout (SBO) events:* If both units are in SBO, various local actions (e.g., actions away from the MCR) are expected to be ongoing during the same time window. It is possible that these actions may impact each other. For example, if an extended loss of alternating current (AC) power (ELAP) is declared at both units, even if the FLEX building housing the equipment for both units is not damaged, it would be a single point of focus for both crews for access and for moving equipment. This could affect crew performance, though it would be challenging to quantify the actual impact.

- *Failures affecting the common low voltage switchyard (and the high voltage switchyard):* Since both units share the switchyards, equipment failures due to hazards (like seismic events, wind-events, external flooding, LOOPWR, or even switchyard fires) may be impacted both by the hazard and due to proximity. In most cases, such failures would be represented in LOOP initiating event frequencies and AC recovery probabilities.

APPENDIX H
COUPLING FACTORS FOR LEVEL 1 MULTI-UNIT RISK

Coupling factors were used in the L3PRA project's integrated site risk (ISR) task to estimate multi-unit core damage frequency (MUCDF) and multi-source risk. For example, coupling factors are used to address identified dependencies between Units 1 and 2 on the reference site.

Previous appendices identified potential cross-source dependencies, including multi-unit (MU) common-cause failures (CCFs), potential inter-unit operator dependencies, and MU seismic correlations. In all cases, coupling factors are needed to calculate or estimate multi-source risk.

This appendix discusses coupling factors used in the L3PRA's MU risk calculations. Section H.1 addresses coupling factors for MU CCFs.[78] This section includes CCF initiating event (IE) frequencies that were modeled in the reactor single unit PRAs. In addition, potential cross-unit CCF coupling factors developed using an alternate approach are presented.

Section H.2 provides a summary of the coupling factors used to estimate MUCDF. Section 6 of the main report describes the MUCDF estimations that were made using the coupling factors presented in this appendix. Appendix I provides some additional information supporting the development of the MUCDF results that are described in Section 6.

H.1 <u>Multi-Unit CCFs</u>

Basic events (BEs) in the single unit PRA were identified in Appendix E as candidates for modeling inter-unit or MU CCFs. Both existing CCFs and BEs for single component failures (e.g., turbine-driven auxiliary feedwater [TDAFW] pumps) were identified as candidates for modeling as MU CCFs. In addition, both component failures leading to IEs and those needed for accident response were identified.

The sections below describe the development of coupling factors for all cases of MU CCFs addressed by the ISR task. Because of how MU risk is estimated in the L3PRA project's ISR task, coupling factors represent the conditional probability of the failure of a Unit 2 component(s), given failure of the identical Unit 1 component(s).

Section H.1.1 provides a brief discussion of the state-of-the-art for CCF modeling, including expansion of CCF group sizes to support MU risk calculations. MU CCFs based on existing CCFs are addressed in Section H.1.2, those that are in new groups are addressed in Section H.1.3, and those for IE frequencies are addressed in Section H.1.4. Some additional notes and rules developed for the MUCDF calculations are provided in Section H.1.5. Section H.1.6 discusses an alternate approach for addressing MU CCFs.

H.1.1 Current State-of-the-Art for CCF Modeling for Large Component Groups

How to address modeling cross-unit MU CCFs, including the issue of large component CCF groups, was included in the L3PRA project's research on understanding the state-of-the-art for

[78] At present, no identical or shared components between the SFPs and reactors have been identified. Consequently, this appendix only discusses cross-unit CCFs.

MUPRA. Overall, the project team determined that there is no current state-of-practice method to evaluate cross-unit or multi-unit CCFs.

The most recent IAEA report on MUPRA (see Section 4.4.5.3 in IAEA [2021a]) recommends "…using a simplified and conservative CCF [factor]" then performing "detailed CCF modeling" for risk-significant inter-unit CCFs. Section 5.2 in the EPRI report on MUPRA (EPRI, 2021a) also suggests a simplified approach and provides some example, generic inter-unit CCF factors.

What the EPRI report recognizes is that there is insufficient data to support detailed CCF modeling for larger component group sizes. Even when CCF data has been collected internationally (see, for example, NEA/CSNI [2022]), CCF data for large component groups is sparse. This viewpoint is supported by the results of an L3PRA project meeting of NRC/RES PRA and data analysis experts and their contractors at Idaho National Laboratory.[79]

As a result, the approach for addressing inter-unit CCFs in the L3PRA project's ISR task is to: (1) use generic and conservative inter-unit CCF factors, and (2) when possible, inform the selection of these factors by the existing single unit CCF factors used by the NRC.

An alternative approach for developing CCF factors using expert judgment and an operational perspective was also performed. The results of this alternative approach are documented in Section H.1.5. These results were not used in the ISR task but the approach is a candidate for future ISR work.

H.1.2 Multi-Unit CCFs Based on Existing CCFs

CCFs for various structures, systems, and components (SSCs) appear in single unit (Unit 1) CDF cutsets. For CCFs that are already modeled in the single unit base PRA, there are two cases: (1) the existing CCF group size is also appropriate for an MU risk model, or (2) the existing CCF group size has to expanded for the MU risk model. For example, if there are CCFs already modeled for a system that is shared by both reactor units (e.g., CCFs of service water pumps) and the success criteria is not changed when going from the single unit model to the MU model, then no expansion of the common cause component group (CCCG) is needed. However, if the CCFs in the single unit reactor PRA model is not in a shared system (e.g., CCF of emergency diesel generators [EDGs]), the CCCG would need to be expanded to address the combined set of components in both reactor units and new CCF parameters would need to be estimated. This latter case is addressed in Section I.1.3.

For an MU event or scenario, the impact of CCFs on the SSCs in both units can be estimated and inserted into the two-unit or MUCDF model. For the ISR task, generic CCF coupling factors were developed using, or informed by, CCF factors available in NRC's SPAR models (NRC, 2021) and the SAPHIRE PRA software (INL, 2011).

First, new CCF group sizes for identical SSCs that appear in MU cutsets are determined. By reviewing the single unit cutsets and using the results of the identification of identical components for the two units (see Section E.2), all identified, identical SSCs from both units were defined as a new CCF group (i.e., the total number of components over both units). For

[79] In addition, an informal communication from an industry PRA expert to the members of the ANS/ASME Multi-Unit PRA Standards Working group states that recent trends in component failure data are making calculations of CCF factors for the *existing* single unit CCFs difficult (i.e., there is less component common-cause failure data now than in previous years).

example, a MU CCF group size for EDGs is four, whereas the single unit (Unit 1) EDG CCF group size is two.

The SAPHIRE software was then used to generate new CCF factors (i.e., alpha factors) (see INL [2012] for discussion of the CCF approach used in SAPHIRE) for the newly postulated CCF group sizes for different component types and failure modes. Using SAPHIRE in this way, example results for two-unit CCF BE probabilities for components that appear often in the single unit cutsets are shown in Table H-1.

Table H-1 also shows the calculated conditional probability of a CCF for Unit 2, given a CCF for the same components and failure mode in Unit 1, that is, the coupling factor (defined as "CCF 4of4" divided by "CCF 2of2"). However, as shown in Table H-1, for simplicity a generic coupling factor of 0.2 is recommended for all the example components and associated failure modes. This type of CCF coupling factor was labeled "CCF1." As Table H-1 shows, a coupling factor of 0.2 is generally conservative for the CCF groups considered in the L3PRA project's ISR task.

Table H-1 Examples of CCF Probabilities Calculated by SAPHIRE for Single Unit and Multi-Unit Group Sizes

Basic Event	**CCF Group Size**	**EDGs** Fail to Start (staggered testing)	**RAT Input Breakers** Fail to Open (non-staggered testing)	**EDG Load Sequencers** Fail to Operate (staggered testing)	**EDGs** Fail to Run (staggered testing)
SAPHIRE CCF BE 2of2	2	3.68E-05	3.50E-04	2.15E-04	3.24E-04
SAPHIRE CCF BE 4of4	4	4.83E-06	8.05E-05	1.55E-05	4.65E-05
CCF 4of4 / CCF 2of2		0.13	0.23	0.07	0.14
Generic CCF1 coupling factor		**0.20**	**0.20**	**0.20**	**0.20**

CCFs involving the nuclear service component water (NSCW) system pumps were considered special cases of potential MU CCFs, given the very large CCCG size. As discussed in Section H.1.5, there were different opinions among experts on whether Unit 2 failures would be considered more or less likely, given Unit 1 failures. For the MUCDF estimations made in the ISR task, these types of MU CCFs were labeled "CCF2" and complete dependence between Unit 1 and Unit 2 failures was assumed (i.e., the generic CCF2 coupling factor was assigned as 1.0). The L3PRA project team judges this coupling factor of 1.0 to be conservative.

The complete list of MU CCF coupling factors used in the ISR task is given in Section H.2.

H.1.3 Multi-Unit CCFs That Are New Groups

As identified in Appendix E, there is only one component type that needs to be represented in MUCDF estimates that was not part of an existing CCF group. Namely, each unit on the reference site has one TDAFW pump. Consequently, the L3PRA project's single unit (i.e., Unit 1) Level 1 PRA model includes BEs for one TDAFW pump. Table H-2 shows the cross-unit CCF probabilities for such events.

Table H-2 New MU TDAFW Pump Basic Events and Combinations

Name	Description	Probability	Uncertainty Distribution
M-TDP-CCF-FTS	CCF of 2 TDPs FTS	1.22E-04	CNI
M-TDP-CCF-FTR	CCF of 2 TDPs FTR	7.79E-04	CNI
M-TDP-FTS	2 TDPs FTS random	3.52E-05	CNI
M-TDP-FTR	2 TDPs FTR random	1.45E-03	CNI
M-TDP-FTS-FTR-12	TDPs FTS1 and FTR2 random	2.25E-04	CNI
M-TDP-FTR-FTS-12	TDPs FTR1 and FTS2 random	2.25E-04	CNI
M-TDP-BOTH-COMB	Sum =	2.84E-03	CNI

For the ISR task, MUCDF estimates should account for both TDAFW pumps, including CCFs involving both pumps and random failures of both pumps. Calculations using SAPHIRE were performed to directly calculate the probabilities of such combinations. Because the corresponding coupling factors would be smaller than the coupling factor of 0.2 used for existing CCFs, the project decided to use the same 0.2 coupling factor for inter-unit TDAFW pump CCFs.

H.1.4 Multi-Unit CCFs For IE Frequencies

For the MUCDF results presented in this report, the only IE that was addressed for MU CCFs in the development of the MUIE frequency is the loss of NSCW. Consistent with the approach used for inter-unit CCFs for the NSCW system, the project team assumed complete dependence between the units with respect to the occurrence of a loss of NSCW. Consequently, the MUIE frequency is assumed to be equivalent to the single unit IE frequency.

Two other types of potential MUIEs were identified in the sitewide dependency assessment (see Appendix C, Section C.2.1.1):

- Interfacing systems loss-of-coolant accident (ISLOCA) from residual heat removal (RHR) cold leg injection lines (two IEs)
- ISLOCA from RHR hot leg injection lines

Due to limited resources, these MUIEs were not addressed by the ISR task. For example, as shown in Table C-1, expert elicitation was used to develop the frequencies for these SUIEs, and a similar effort would be needed to develop frequencies for the associated MUIEs. Also, both IEs represent less than 1 percent of SUCDF. Since cross-unit dependencies are expected to be weaker across units (as opposed to within a unit), the contribution to MUCDF also would be expected to be small.

H.1.5 Additional Notes and Rules Developed for MUCDF Calculations

During the process of assigning coupling factors for CCF failures appearing in various hazard category CDF models, the following additional assignments were made:

- For three- and four-element cutsets containing CCF basic events, a generic BE coupling factor of 0.20 is assigned (except for cutsets containing NSCW CCF basic events with a very large CCCG size, for which a coupling factor of 1.0. is assigned). The factor of 0.2 is chosen since it is deemed to bound most SSC CCFs.

- If a cutset has two BEs, an IE frequency and a CCF, a BE coupling factor of 0.2 is assigned.

- In some cutsets with 3 BEs, the third BE (non-CCF) may be left as is (classification-wise); this results in a default coupling factor of 1.0 for that BE.

- For cutsets containing TDAFW FTS and FTR random failures, a factor of 0.2 is assigned per analyst judgement.

H.1.6 Alternate Approach for Addressing Multi-Unit CCFs

As noted above, there is no current state-of-practice method to evaluate cross-unit CCFs. Therefore, another evaluation using analyst judgment was performed and is provided as an alternative approach to that described in previous sections in this appendix. This alternative approach addresses some potential inter-unit CCFs that were not addressed above.

Section H.1.6.1 addresses cross-unit CCFs for components that exist as part of CCCGs in the single unit PRA, while Section H.1.6.2 addresses cross-unit CCFs for types of components where only a single component of that type exists in each unit. Section H.1.6.3 addresses MU CCFs that can contribute to MUIEs.

H.1.6.1 MU CCFs for Mitigating Components

The sitewide dependency evaluation identified many mitigating system CCFs that met the ASME/ANS PRA standard for significant events. The following components have associated CCF events with a Fussell-Vesely (FV) importance measure greater than or equal to 0.005:

- reserve auxiliary transformer (RAT) breakers
- EDG load sequencers
- EDGs
- AFW pumps
- EDG fuel oil transfer pumps

In addition to these components, the number of significant events that have a risk achievement worth (RAW) importance measure of 2 or more is extensive. As expected, this list is dominated by CCF events that have low probabilities, but their failure could result in significant risk increases. The following is a partial list of systems with components that are part of significant CCF events:

- NSCW system—pumps (not initiating event), spray valves and associated I&C components, fans

- DC power system—battery chargers, batteries, inverters

- Reactor protection system (RPS)—control rods, reactor trip breakers, bi-stables, logic modules
- AFW system— pump suction and discharge check valves, line check valves, etc.
- Emergency AC power system—relays, fuel transfer components, EDGs, room vents or dampers

Given these results, the components associated with CCF events that were identified as significant based on their FV importance were evaluated to determine their potential for cross-unit CCF. The components associated with CCF events that have the highest RAW values were also evaluated. Note that the evaluation provided in the following sections has grouped, where possible, components whose CCF would result in the same loss of function.

RAT Breakers and EDG Load Sequencers

CCFs of the RAT breakers and EDG load sequencers are the largest component failure contributors to the single unit CDF. One consequence of EDG load sequencer failure is that the RAT breakers will fail to open. In addition, the individual failure and CCF probabilities are similar between these two components. Therefore, a single evaluation of the potential cross-unit CCF of these two components was performed focusing on the failure of all four site RAT breakers to open.

Determining a reasonable conditional CCF of RAT breakers at the second unit given the CCF of both RAT breakers at the first unit is a bit more difficult than estimating the conditional CCF for the expansion of a large CCCG because the CCF failure mechanism is only assumed to be present in two components at the single unit. This is very different, for example, than the MU CCF of the NSCW pumps where six pumps failed at the first unit, indicating the CCF mechanism is more widespread and, therefore, more likely to be experienced at the second unit. However, there is likely greater CCF potential for the RAT breakers because they are not frequently tested or operated. Therefore, a latent CCF mechanism could be in place and overlap (timewise) with the other unit.

Given these considerations, a smaller, but likely still conservative, conditional CCF probability of 0.05 was selected for the RAT breakers at the second unit given CCF of the RAT breakers at the first unit. This conditional CCF probability could be evaluated via parametric sensitivity analyses to determine the impact of this uncertainty, though such analyses were not performed as part of the L3PRA project.

Emergency AC Power Components

The CCF of various emergency AC power components are significant risk contributors to the single unit model. And with a sitewide LOOP being the most likely MUIE, the emergency AC power components are important to evaluating MU and multi-source risk. The CCF of the EDGs to run is the most risk-significant CCF in terms of its FV and RAW importance measures. Therefore, the evaluation focuses on the EDGs themselves but includes considerations outside of their component boundaries that could result in a sitewide loss of emergency AC power.

A CCF of both EDGs on a single unit does not necessarily mean a widespread CCF mechanism exists. However, the EDGs are run for only short periods during testing. In addition, they are not run in the same fashion as when following an actual demand. Therefore, there is greater potential for a latent CCF mechanism to be present on all four EDGs as compared to continually running components.

Given these considerations, a 0.1 conditional CCF probability was selected for the EDGs at the second unit given CCF of the EDGs at the first unit. This probability is likely conservative and could be evaluated via parametric sensitivity analyses to determine the impact of this uncertainty, though such analyses were not performed as part of the L3PRA project. Note that a lower conditional CCF probability would result in the random failure of all four site EDGs to run becoming a more significant contributor to sitewide risk than the MU CCF of the EDGs.

AFW System Components

The dominant CCF for AFW system components is the CCF of all three AFW pumps to run. Since the AFW pumps have different drivers (two motor-driven pumps and a single turbine-driven pump), this CCF event only covers the volute portion of the pumps that are similar. The CCFs of various AFW system valves (mostly check valves) have significant RAW importance measures but insignificant FV values due to their low failure probabilities. Therefore, the focus of this evaluation is on the CCF of the AFW pumps to run.

The dominant cutsets for this CCF event involve a transient (e.g., reactor trip, loss of main feedwater, or loss of condenser heat sink) with the subsequent failure of operators to restart main feedwater (if available) and to initiate feed and bleed cooling. The likelihood of a dual-unit transient of this nature is very low. Even if these initiating events did occur within the same PRA mission time, operators for the second unit would have additional information on the CCF of the AFW pumps and the actions needed to mitigate the loss of decay heat removal, which would decrease the likelihood of failure.

Given these considerations, the likelihood of dual-unit CCF of the AFW pumps to run is assumed to be sufficiently low as to not require a detailed risk calculation and is screened out from further evaluation. Note that the CCF of both unit's turbine-driven AFW (TDAFW) pump during a sitewide LOOP could be risk significant and is evaluated in Section H.1.6.2.

NSCW System Components

The evaluation of the sitewide loss of NSCW initiating event is provided in Section H.1.6.3, while the evaluation provided in this section focuses on the dual-unit loss of NSCW given an unrelated initiating event. The CCF of the various NSCW components, most notably the NSCW pumps and components associated with the cooling tower spray valves, have high RAW importance measures but low FV values due to their low probabilities. The CCF probability of the NSCW pumps to run on single unit given an initiating event (8×10^{-8}) is low. Operators have the ability to reduce heat loads and maintain the ultimate heat sink by aligning for single pump operation for partial CCFs of the NSCW pumps. Therefore, it is judged that a loss of both unit's spray capability has the largest potential for sitewide loss of NSCW given an initiating event and should be the focus of this evaluation.

The CCF of the spray valves to open is only applicable for initiating events that result in the spray valves closing (e.g., LOOP) or if the spray was being bypassed before the initiating event but is needed within 24 hours to mitigate the event. This latter failure is likely to be slower moving and, therefore, a lesser multi-unit concern. In addition to the CCF of the spray valves failing to open, the CCF of the spray valves to close during a LOOP is also a concern. A water hammer event can occur if the spray valves remain open when the NSCW pumps restart after being sequenced onto the EDGs. While a CCF of both spray valves to open or close does not indicate a widespread CCF mechanism exists, the spray valves are not operated often and, therefore, the potential for a latent CCF mechanism that could be present on all four NSCW spray valves is more likely than for continually running components.

Given these considerations, a 0.1 conditional CCF probability is applied to the other unit's NSCW spray valves given the CCF of both spray valves at one unit. This probability is likely conservative and could be evaluated via parametric sensitivity analyses to determine the impact of this uncertainty, though such analyses were not performed as part of the L3PRA project.

DC Power System Components

The CCF of the safety-related batteries, battery charger, and inverters have significant RAW values given their low probabilities and the risk impact of a loss of all safety-related DC power. The most likely CCF is the battery chargers and, therefore, is the focus of this evaluation.

While a CCF of all four battery chargers for safety-related buses 'A' and 'B' shows that a potential widespread CCF mechanism may exist, the continually powered nature of these components would likely mean different timing of failures, which would likely decrease the CCF potential of the battery chargers of the other unit during the same period.

Given these considerations, a conditional CCF probability of 0.05 for the battery chargers at the other unit was selected. This conditional CCF probability could be evaluated via parametric sensitivity analyses to determine the impact of this uncertainty, though such analyses were not performed as part of the L3PRA project.

RPS System Components

The CCF of multiple RPS components (control rods, reactor trip breaker, etc.) have significant RAW values given their low probabilities and the risk impact of an ATWS. However, the single unit likelihood of an ATWS and its corresponding CDF are low. A dual-unit ATWS is likely a very low risk event due to several factors. First, a dual-unit ATWS would require the need for a reactor trip at both plants, with the most likely scenario being a sitewide LOOP. However, ATWS risk is very low for LOOPs because the only applicable RPS CCF is the control rods. Second, for scenarios other than a CCF of the control rods, operators would have information about potential RPS failures that could be present in the RPS at the redundant unit. This could potentially increase the likelihood that operators manually trip the reactor prior to an RCS pressure excursion. Third, the likelihood of the CCF of the control rods at both units is likely to be extremely low. Control rod movement is verified per a monthly TS surveillance, which would reduce the likelihood for a sufficient number of rods to be stuck to prevent a reactor shutdown.

Given these considerations, the potential for dual-unit ATWS from CCF of the RPS components is assumed to be sufficiently low so as to not require a detailed risk calculation and is screened out from further evaluation.

H.1.6.2 Potential MUCCFs for Single Component Failures Model in Single Unit PRA

In addition to CCFs that could fail components at both units, single component failures on one unit could result in the failure of the identical components on the other unit via a CCF mechanism. A review of significant events identified the normal charging pump (NCP) and TDAFW pump as single component failures that could result in multi-unit risk concern if the identical component of the other unit failed either due to random failure or common cause.[80]

Since the failure of the individual components does not represent a CCF, the use of existing CCF parameters can be used as bounding conditional CCF probabilities, given a failure of the component in one unit, for the identical component in the other unit. Specifically, the conditional CCF probabilities for the NCP or TDAFW pump in one unit can be estimated using the α2 values given the failure of either of these components in the other unit. Using the α2 values in this manner (i.e., cross-unit CCF) is believed to be conservative given the mitigating factors already discussed in this report. The recommended conditional CCF probabilities for the NCP and TDAFW pump are provided below.

Normal Charging Pump

The mean α2 values for normally running motor-driven pumps (MDPs) for a clean water system are provided in the table below.

Table H-3 Mean α2 Values for Normally Running MD Pumps

Component and Failure Mode	Mean α_2
CLN-MDP-NR-FS	0.00617
CLN-MDP-NR-FR	0.0126

Therefore, the conditional CCF probability of the NCP on the redundant unit given the failure of the NCP on the other unit can be estimated to be 0.019 (accounting for both failure to start and failure to run).

Turbine-Drive Auxiliary Feedwater Pump

The mean α2 values for TDAFW pumps are provided in the table below.

Table H-4 Mean α2 Values for TDAFW Pumps

Component and Failure Mode	Mean α_2
AFW-TDP-FS	0.0205
AFW-TDP-FH (<1 hour)	0.0205
AFW-TDP-RH (1–24 hours)	0.0196

[80] CCF was not considered in single unit PRA model for certain components (e.g., electrical buses and the condensate storage tank) because it was beyond the state-of-practice and, therefore, these components were not evaluated for potential cross-unit CCF.

Because there are two α2 values for the failure to run (FTR), which are divided into early (i.e., first hour of operation) and late (i.e., the remaining 23 hours of operation) terms, an effective α2 was calculated for the for the overall FTR using the 24-hour random and CCF probabilities. For the TDAFW pumps, this effective α2 is calculated to be 0.020. Therefore, the conditional CCF probability of the TDAFW pump on the redundant unit given the failure of the NCP on the other unit can be estimated to be 0.04 (accounting for both failure to start and failure to run).

H.1.6.3 MU CCFs for Initiating Events

As part of the Phase 1 sitewide dependency assessment for cross-unit initiating events (see Appendix C), the CCFs of the NSCW pumps and the RHR system isolation valves were identified as potential risk-significant initiating events that could concurrently affect both units.[81] The CCF of all NSCW pumps would result in core damage if the reactor coolant pump (RCP) seals fail. In addition, the CCF of the RHR system isolation valves could result in an ISLOCA that eventually would result in core damage. A discussion of the evaluation of the potential for cross-unit CCF resulting in these concurrent core damage events at both units is provided below.

Sitewide Loss of NSCW Initiating Event

A loss of NSCW initiating event is assumed to be the loss of adequate flow from both trains. Specifically, the loss of flow from at least 2 out of 3 pumps in both trains. The most risk-significant failure is the CCF of all 6 pumps; however, CCF of 4 or 5 pumps is also considered in the L3PRA model.[82] If at least one NSCW pump is available, operators are directed to the trip the reactor, trip the RCPs, and isolate chemical and volume control system (CVCS) letdown. Operators can then place at least one NSCW train in single pump operation, which is sufficient to prevent a challenge to the RCP seals.

There is potential for CCF of the NSCW pumps of both units (12 total) because the pumps share the same CCF coupling factors (design, maintenance, operation, environment, etc.). However, the expansion of the NSCW pump common-cause component group (CCCG) to account for all 12 pumps is not appropriate because the CCF data is not collected across systems or units.[83] Therefore, it is not known how strong the CCF coupling is across units. There are mitigating factors for the potential cross-unit CCF. The most notable involves timing of potential CCFs. For example, design changes and maintenance are typically staggered across units. Therefore, it is likely that CCFs between the two units would not occur within the same PRA mission time if the same CCF mechanism is present. In addition, at least two NSCW pumps in each train are continually running, which would increase the likelihood that a potential CCF mechanism would be identified before being introduced or occurring on the other unit.

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[81] The risk-significant CCFs were identified using the ASME/ANS PRA Standard definition for significant events—basic events with a Fussell-Vesely (FV) importance measure greater than or equal to 0.005 or risk achievement worth (RAW) importance measure of 2 or more.

[82] A loss of NSCW initiating event is not assumed to occur given the CCF of the NSCW cooling tower fans or sprays because a slower system heat-up would occur in these scenarios, which would result in a technical specification (TS) directed shutdown instead of a reactor trip.

[83] There are considerable uncertainties associated with the CCFs of CCCGs sizes of 4 or more because most complete CCF events in the CCF database are for CCCG sizes of 2 or 3.

Based on work associated with the development of the Causal Alpha Factor Method (CAFM), the CCF data show that the strongest CCF coupling for most components is a shared environment. This especially applies to service water systems, where operating experience has shown biologic and other natural phenomenon resulting in CCF of service water pumps and plugging of traveling screens or strainers. However, environmental CCF of the NSCW pumps at the reference plant is mitigated because the system is a semi-closed system that is chemically treated.

Therefore, the potential cross-unit CCF of the NSCW pumps is likely low. However, a sitewide loss of NSCW cannot be ruled out. First, although mitigated, the CCF of all 12 NSCW pumps is possible. This CCF potential is likely to be from the standby portion of the NSCW system (e.g., design change of instrumentation and control (I&C) associated with the pumps). And although the likelihood of an environmental CCF of the NSCW system is decreased, it cannot be ruled out (e.g., incorrect treatment results in corrosion of system piping).

Given these considerations and based on expert judgment, a 0.1 conditional CCF probability is applied to the other unit's NSCW pumps given the CCF of all six pumps at one unit.

Concurrent ISLOCAs Due to Failure of RHR System Isolation Valves

The risk significant ISLOCA is from failure of the RHR system hot leg suction isolation motor-operated valves (MOVs). If both MOVs fail in either hot leg 1 or 4, an unisolable ISLOCA would occur that eventually results in core damage due to loss of inventory outside containment. The potential CCF of the two isolation valves in the RHR system hot leg suction lines was not modeled using the alpha factor method. Instead, expert elicitation was used to determine the failure probability of the isolation MOV that normally experiences reactor coolant system (RCS) pressure and the conditional failure probability of the redundant MOV. The expert elicitation did not consider the potential CCF of identical MOVs between the two units.

The most likely potential for CCF is between the isolation MOVs that experience RCS pressure on both units.[84] As these MOVs share the same CCF coupling factors and are normally closed and do not change position, the potential for this CCF is judged to be more likely (compared to NSCW pumps). However, this case does not include a traditional CCF of the first unit and, therefore, does not actually have a CCF mechanism assumed in the single unit cutset.

Using the same, and likely very conservative, NSCW pump conditional CCF probability of 0.1 for the redundant RHR isolation valve that experiences RCS pressure (in conjunction with the failure probabilities identified in the expert elicitation) results in a point estimate multi-unit ISLOCA frequency of 2×10^{-11} per reactor-critical-year.[85] This conditional CCF probability could be evaluated via parametric sensitivity analyses to determine the impact of this uncertainty, though such an analysis was not performed as part of the L3PRA project.

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[84] The conditional failure probabilities are likely similar for redundant MOVs once the first valve in series fails because the resulting pressure pulse is the most likely failure cause of the second valve in series.

[85] There is also a random failure contribution; however, it is over three orders of magnitude lower than this frequency and, therefore, is likely to be a negligible risk contributor.

H.2 Summary of Coupling Factors Used to Estimate MUCDF

The overall approach for the assignment of coupling factors to be used in MUCDF estimates is intended to be simple and conservative. Table H-5 and Table H-6 show the coupling factors used in the L3PRA project's ISR task to estimate MUCDF. These tables illustrate the simplicity of the approach by the limited number of unique coupling factors used. The previous sections discussed the conservativism in individual coupling factor assignments.

Table H-5 and Table H-6 show the generic coupling factors and all assigned coupling factors, respectively, by cutset type. Cutset type assignments are made during the review of single unit (Unit 1) Level 1 PRA cutsets described in Section 6. In particular, the cutset type is associated with the inter-unit dependency considered to be the most important to MUCDF estimation. For internal events MUCDF, the cutsets containing inter-unit CCFs are considered the most important and are assigned to either type CCF1 or CCF2. Cutsets for other hazards (e.g., high winds or seismic events) are assigned to either type STRUCTURE or SEISMIC.

Inter-unit dependencies for operator actions are treated as a secondary contribution to the overall dependencies between the two reactors. Example MUCDF calculations confirmed that this approach is appropriate for the plant-specific, Level 1 internal events results for the reference plant. Due to how MUCDF is estimated for the ISR task (see Section 6 and Appendix I for discussion), it is conservative to ignore the effect of operator actions in cutsets that have been assigned a cutset type (e.g., CCF1 or CCF2) because the approach assumes all other BEs in the MU cutset have a failure probability of 1.0.[86]

The RANDOM cutset type has only random BEs (i.e., there are no inter-unit dependencies that need to be addressed).

Table H-5 Generic Coupling Factors by Cutset Type

Cutset Type	Coupling Factor	Applicability	Notes
STRUCTURE	1.0	Certain hazards such as seismic events	
SEISMIC	1.0	Seismic events	
CCF1	0.2	All CCFs except those for NSCW	
CCF2	1.0	Certain NSCW components and failure modes	
RANDOM	None	Cutsets with only random BEs	
OTHER			

[86] However, as noted in Section 6.2.3.2, a sensitivity analysis for the LOOPWR MUIE demonstrated that a more rigorous and complete assignment of coupling factors would not change the estimated MUCDF appreciably.

Table H-6 All BE Coupling Factors

I.	**"Generic" U2 Coupling Factors by Cutset Type**		
	Cutset Type	**Coupling Factor**	
	STRUCTURE	1	direct CD; 2 element cutsets
	SEISMIC	1	only one seismic failure BE
	CCF1	0.2	For all SSC but NSW
	CCF2	1	Only for NSW
	HEP	0.1	(If HEP <0.10.) Customize as needed.
	OTHER	0.01 – 0.2	Customize as needed.
	RANDOM	scenario specific. Use U1 CCDP.	
	CCF1-SEISL	0.2	Seismically induced LOOP and a CCF1
	CCF1-2	0.04	0.20 * 0.20
	CCF1-HEP	0.02	0.20 * 0.10
	CCF2-HEP	0.1	1.0 * 0.10
II.	**Additional U2 Coupling Factors - Cutset Level**		
	(use is optional at the discretion of the analyst)		**U1 basic event name**
	CCF RAT Input Breaker Fail to Operate	0.23	1-ACP-CRB-CF-A205301
	CCF Load Sequencers Fail to Operate	0.07	1-EPS-SEQ-CF-FOAB
	CCF EDGs FTR	0.14	1-EPS-DGN-CF-FRUN1
	CCF EDGs FTS	0.13	1-EPS-DGN-CF-FSUN1
	CCF AFW-TDP	0.2	New MU basic events.
	HEPs - total dependence	1	
	HEPs - high dependence	0.5	
III.	**Level of Seismic Correlation**	**Factor**	
	No Correlation	0	
	Weak Correlation	0.2	
	Moderate Correlation	0.5	
	Strong Correlation	0.8	
	Full Correlation	1	
	(*) = CCF SWS MDP and MOV: 4 or more combinations		
IV.	**Seismic Cutset Combination of Basic Events (not all combinations are shown)**		
	SEISMIC2	1	2 seismic BE failures in the same cutsets
	SEISMIC2R	0.01-0.10	2 seismic failure Bes + 1 RCP seal BE
	SEISMIC2U1	0.01-0.10	2 seismic BEs; 1 is seismically uncorrelated.
	SEISMIC3	1	3 seismic failure basic events
	SEISMIC3R	0.01-0.10	3 seismic events + 1 RCP seal BE
	SEISMIC3U1	0.01-0.10	3 seismic failure basic events; 1 is uncorrelated
	SEISMIC4U1R	0.01-0.10	4 seismic events (1 is uncorrelated) + 1 RCP seal BE
	SEISMICR	0.01-0.10	1 seismic BE event and 1 RCP seal BE
	SEISMICU2	0.01-0.10	2 uncorrelated seismic BEs

Table H-6 All BE Coupling Factors (cont.)

	Cutset Type	Coupling Factor	
	SEISMIC2L	1	2 seismic BEs; 1 is seismically induced LOOP
	Additional cutset types, as they were later observed, are added, as shown below. Their descriptions, if not listed, can be inferred from the ones listed above.		
	SEISMIC2LHEP	0.01-0.10	
	SEISMIC2LR	0.01-0.10	
	SEISMIC2U1R	0.01-0.10	
	SEISMIC3L	0.01-0.10	
	SEISMIC3RL	0.01-0.10	
	SEISMIC3U1L	0.01-0.10	
	SEISMIC3U2L	0.01-0.10	
	SEISMICht	1	1 seismic failure BE and "FLEX failure" BEs

APPENDIX I
ESTIMATE MULTI-UNIT LEVEL 1 RISK

This appendix provides additional information regarding the calculation of multi-unit core damage frequency (MUCDF) using the cutset estimation method (CEM) approach described in Section 6 of the main report. Section I.1 provides illustrative examples of certain aspects or steps of the CEM approach for calculating MUCDF. Section I.2 identifies some specific omissions in calculation MUCDF using the CEM approach. Section I.3 summarizes the MUCDF results that have been developed for the ISR task using the CEM approach, including results for a “FLEX sensitivity case” that addresses FLEX strategies[87] and some other plant updates (e.g., new RCP seals).

I.1 Illustrative Examples of MUCDF Calculations

This section provides illustrative examples of certain aspects or steps of the CEM approach for calculating MUCDF. Each successive example addresses additional aspects or steps. The application of the CEM approach for the MUIEs in the first three examples below only involved a single (initial) cutset review. The fourth example (for LOOPWRs) illustrates application of multiple cutset reviews.

I.1.1 Example: Grid-Related LOOPs

The first example illustrates the types of results developed by the CEM approach for grid-related losses of offsite power (LOOPGRs). Table I-1 summarizes the relevant terms and inputs from the Unit 1 PRA (or SUPRA) and calculated terms and final MUCDF results for LOOPGRs. From the SUPRA, the following inputs are needed:

U1IEF	U1 initiating event frequency
U1CDF	U1 core damage frequency
U1-CCDP	U1 conditional core damage probability
N	number of cutsets selected to represent U1CDF in MUCDF calculations
M	total number of cutsets generated for LOOPGR
U1CDF(M)	U1CDF for all M cutsets
U1-CDF(N)	U1-CDF for N cutsets (determined from U1 PRA results for LOOPGR)

From the results shown in Table I-1, it is observed that MUCDF for LOOPGRs is 5.5 percent of the U1CDF for LOOPGRs.

Using the U1 PRA inputs identified above and the MUIE frequency for LOOPGRs given in Appendix C, the maximum possible and minimum possible MUCDFs can be calculated directly.

[87] 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 offsite.

The MAX-MUCDF assumes that, if U1 undergoes a core damage event, U2 also will go to core damage (i.e., there is total coupling between the two units). MIN-MUCDF assumes that the two units are not coupled; that is, if U1 undergoes a core damage event, U2 may randomly undergo a coincidental core damage event with a conditional core damage probability equal to that of U1. The bar chart in Section 6 of the main report (Figure 6-1) confirms that the MUCDF values calculated using the CEM approach are between MIN-MUCDF and MAX-MUCDF calculated values.

Also, the U1 PRA inputs identified above can be used to calculate the scale-up factor that is needed to adjust the MUCDF obtained using the CEM approach to account for only using 95 percent of the U1CDF.

Table I-1 Illustrative Example of CEM Results for a MU-LOOPGR Scenario

Input/Result	Definition	LOOPGR Numerical Values	Terms and Equations Used for Calculations
U1IEF	U1 IE frequency	1.23E-02/rcy	f
U1CDF	U1 CDF	1.83E-05/rcy	g
U1-CCDP	U1 conditional core damage probability	1.49E-03	c = g / f
MUIEF	MU IE frequency	6.15E-03/rcy	d
MUCDF	MU CDF (calculated)	**1.00E-06/rcy**	e (estimate by CEM)
MU-CCDP	MU conditional core damage probability	1.63E-04	e / d
MAX-MUCDF	Maximum possible MUCDF	9.16E-06/rcy	d × c × 1.0
MIN-MUCDF	Minimum possible MUCDF	1.37E-08/rcy	d × c × c
N	Number of cutsets selected to represent U1 CDF for CEM approach	409	selected
M	Total number of cutsets for U1 LOOPGR CDF results	24008	from U1
U1CDF(N)	U1 CDF represented by N cutsets	1.74E-05/rcy	a
U1CDF(M)	U1 CDF represented by all M cutsets in SUPRA for LOOPGR	1.83E-05/rcy	b
% CDF(N) / CDF(M)	Scale-up factor used to adjust MUCDF	**95.1%**	a / b

I.1.2 Example: Switchyard-Centered LOOPs

The second example illustrates how the cutset review process was conducted for the CEM using U1 PRA results for switchyard-centered losses of offsite power (LOOPSCs). Table I-2 shows relevant information from the U1CDF LOOPSC results.

Table I-2 U1 LOOPSC PRA Information

Parameter	Value
U1 LOOPSC IEF	1.04E-02/rcy
U1 LOOPSC CDF	1.04E-05/rcy
U1 LOOPSC CCDP	9.95E-04
# of Cutsets	17114
Top 95% Cutsets	133
CDF of Top 95% Cutsets	9.83E-06/rcy

In turn, Table I-3 shows the initial examination of minimal cutsets that was done in Excel spreadsheets to identify CCFs to represent as multi-unit CCFs (MU CCFs). Table I-3 specifically shows the top 10 minimal cutsets for LOOPSC out of a total of 133 minimal cutsets used to represent U1CDF for MUCDF calculations.

During examination of the U1CDF cutsets in preparation for applying the CEM approach, BEs were marked by “coloring” them. In particular, as shown in Table I-3, CCF BEs are highlighted in yellow. In this example, only CCF BEs are “colored.” Next, each cutset is assigned a “cutset type,” using the labeling scheme provided in Table 6-1 of the main report. Those cutsets that contain “random failures” are classified as the “RANDOM” type and may not be explicitly identified in the “cutset type” column (i.e., those cutsets that have a blank type are to be treated as “RANDOM”).

Additional colors were used in other CEM applications (as shown below). It should be noted that the color convention was not necessarily consistent from one hazard category to another.

Once CCF BEs are identified and the cutset type is selected, then a BE coupling factor is assigned, which is illustrated in the next section below.

Table I-3 Top 10 Minimal Cutsets for U1 LOOPSC (NO-FLEX Case) - Cutset Review for MUCDF Calculations

Cutset Type	#	Prob/Freq (/rcy)	Total %	CutSet	Description
	Total	9.83E-06	100	Displaying 133 CutSets. (133 Original)	
CCF1	1	3.64E-06	37.01		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
CCF1	2	2.23E-06	22.72		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE
	3	6.48E-07	6.59		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
	4	2.98E-07	3.03		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
	5	2.48E-07	2.52		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
	6	2.48E-07	2.52		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO

Table I-3 Top 10 Minimal Cutsets for U1 LOOPSC (NO-FLEX Case) - Cutset Review for MUCDF Calculations (cont.)

Cutset Type	#	Prob/Freq (/rcy)	Total %	CutSet	Description
CCF1	7	1.93E-07	1.96		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
	8	1.85E-07	1.88		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
	9	1.85E-07	1.88		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
	10	1.51E-07	1.54		
		1.04E-02		1-IE-LOOPSC	LOSS OF OFFSITE POWER (SWITCHYARD- CENTERED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		2.72E-03		1-DCP-BAT-MA-BD1B____	BATTERY 1BD1B IN MAINTENANCE
	.....				

I.1.3 Example: Plant-Centered LOOPs

The third example further illustrates how the cutset review process was conducted for the CEM approach, as well as showing the assignment of cutset types and associated BE coupling factors. This example uses U1 PRA results for plant-centered losses of offsite power (LOOPPCs). Table I-4 is provided below to help illustrate the following points:

- CCF basic events are highlighted in yellow.
- BEs for random equipment failures are highlighted in green.
- If a U1 cutset is labeled as “RANDOM,” no coupling factor (“a”) is assigned (that column is left blank). The cutset contribution to MUCDF is obtained by multiplying the cutset CDF by only the U1-CCDP (“b”).
- If a cutset is labeled other than “RANDOM,” it is assigned a coupling factor “a” (using the guidance in Section 6.2.3.3 and Table 6-1 of the main report) that is summed with the U1-CCDP (“b”) to calculate the cutset contribution to MUCDF. The rare events approximation is used to calculate the sum (shown as “n” in Table I-4). The calculation of MUCDF for a specific cutset is shown in the far-right column in Table I-4. (Note that the cutset contribution for each cutset has already been adjusted to include the MUIE frequency, rather than the UI-IE frequency in the original U1 cutsets).
- Table I-4 shows that there are four cutsets that contain the same human failure event (HFE) related to restoration of AC power systems after offsite power has been recovered. Based on the results of the sitewide dependency assessment given in Appendix F, these HFEs are considered to be independent (i.e., failure of this operator action in Unit 1 is not related to failure of the same action in Unit 2). The implication of this unmarked basic event depends on the assigned cutset type. For example:
    - For cutset #7 shown in Table I-4, the HFE is not marked and is not considered in coupling factor assignment. This is equivalent to the Unit 1 and Unit 2 HFEs being treated as completely dependent, which is not consistent with the sitewide dependency assessment and, therefore, produces a conservative result. If this cutset (and others like it) made a significant contribution to MUCDF (which it did not), then a cutset coupling factor (rather than just a BE coupling factor) could be assigned, as described in the next example.
    - If the Unit 1 and Unit 2 HFEs were considered to be dependent, then no additional adjustment would be required, as the calculation shown reflects complete dependence between them.
    - The other cutsets shown in Table I-4 that contain this HFE (i.e., cutsets #3, #5, and #6) are all assigned a cutset type of “RANDOM.” For these cutsets, since their contribution to MUCDF is calculated by multiplying the cutset CDF by only the U1-CCDP, no additional adjustment is needed specifically for the HFE.

Table I-4 Selected LOOPPC Cutsets for Limitations Discussion

MUIE-LOOPPC frequency is substituted into the table. Cutset frequencies are recalculated. (All frequencies are in terms of per reactor-critical-year.)								
U1IEF		1.93E-03	This information is for the total LOOPPC IE.					
U1CDF		1.91E-06	Only 7 out of 120 cutsets are shown below for discussion.					
U1-CCDP		9.91E-04	Only U1-CCDP value is used below for calculations					
# Of Cutsets		6686						
Top 95% Cutsets		120						
CDF of Top 95% Cutsets		1.82E-06			MUCDF (for a cutset) = U1 cutset CDF * n			
MUIEF		1.07E-04	This frequency is substituted into the 7 cutsets below.		MUCDF (sum of 9 cutsets only)			1.26E-08
		7.73E-08	= Sum of 9 U1 Cutset CDFs below, with MUIEF value substitution.					
Cutset Type	**#**	**Prob/Freq**	**CutSet**	**Description**	**(a)**	**(b)**	**n = (a+b)-ab**	**MUCDF (*)**
CCF1	1	3.75E-08			0.2	9.91E-04	2.01E-01	7.52E-09
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN				
CCF1	2	2.30E-08			0.2	9.91E-04	2.01E-01	4.62E-09
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		2.15E-04	1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE				

Table I-4 Selected LOOPPC Cutsets for Limitations Discussion (cont.)

RANDOM	3	6.68E-09				9.91E-04	9.91E-04	6.62E-12
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		3.30E-02	1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)				
		3.30E-02	1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)				
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO				
RANDOM	4	3.06E-09				9.91E-04	9.91E-04	3.04E-12
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		5.35E-03	1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN				
		5.35E-03	1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN				
RANDOM	5	2.55E-09				9.91E-04	9.91E-04	2.53E-12
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		3.30E-02	1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)				
		1.26E-02	1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE				
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO				
RANDOM	6	2.55E-09				9.91E-04	9.91E-04	2.53E-12
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		3.30E-02	1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)				
		1.26E-02	1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE				

Table I-4 Selected LOOPPC Cutsets for Limitations Discussion (cont.)

		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO				
CCF1	7	1.99E-09			0.2	9.91E-04	2.01E-01	3.99E-10
		1.07E-04	MU-IE-LOOPPC	MU LOSS OF OFFSITE POWER (PLANT- CENTERED)				
		3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN				
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO				

I.1.4 Example: MUCDF for Weather-Related LOOP

The fourth example illustrates application of the CEM approach using additional cutset review to identify U1 PRA cutsets that are candidates to have cutset coupling probabilities calculated and applied. This example uses the U1 PRA results for weather-related losses of offsite power (LOOPWRs). It should be noted that the results for LOOPWRs given elsewhere are based on only a single pass through the SU cutsets.

Table I-5 shows the input values needed to calculate MUCDF for LOOPWRs. Table I-6 shows the results for MUCDF obtained by applying the CEM approach using additional cutset review. Note that the MUCDF is less than 5 percent of the U1 CDF.

Table I-5 Inputs from Level 1 PRA Results for LOOPWR

	Value	Term
U1 LOOPWR IEF	3.91E-03/rcy	
U1 LOOPWR CDF	9.02E-06/rcy	*a*
U1 LOOPWR CCDP	2.31E-03	
# of Cutsets	14554	
Top 95% Cutsets	315	
CDF of Top 95% Cutsets	8.56E-06/rcy	b
Scale-up factor	1.05E+00	a/b

Table I-6 Results for LOOPWR MUCDF Using CEM Approach

	Value
MUIEF	2.44E-03/rcy
MUCDF based on 95% of U1 cutsets	4.13E-07/rcy
Final MUCDF (after applying scale-up factor of 1.05)	4.35E-07/rcy

Table I-7 through Table I-9 illustrate the steps taken to develop the MUCDF value shown in Table I-6. The information provided in Table I-7 forms the basis for applying the CEM approach. This table lists significant CCFs, HFEs, and other BEs identified through review of LOOPWR cutsets for the U1 PRA model. The following notes pertain to the table:

- All CCFs that are to be modeled as MU CCFs are highlighted in yellow.
- For each CCF, the BE coupling factor to be used is provided, as well as the number of occurrences for each CCF in the 315 cutsets chosen to calculate MUCDF.
- Human failure events that are judged to be independent are highlighted in green.
- AFW turbine-driven pump failures that are to be represented as cross-unit CCFs are highlighted in dark blue along with their associated coupling factors.

- RCP seal failures (which are judged to be independent between U1 and U2) are highlighted in pale green.

- Human failure events that are to be represented as dependent between U1 and U2 are highlighted in pale yellow (with a coupling factor of 1.0).

- NSCW CCF combinations for 4-of-6 and 5-of-6 components should be represented as dependent failures as indicated in the table note shaded in orange.

Table I-8 illustrates the cutset review done for the U1 LOOPWR cutsets. It should be noted that the cutsets shown in Table I-8 do not represent all the cutset types identified in the LOOPWR cutset review. The first seven LOOPWR cutsets are shown with the following points of interest:

- The cutset types shown are either “CCF1,” “RANDOM,” or “RANDOM+HEP.”

- All CCFs are assigned a BE coupling factor.

- BE coupling factors are used for HFEs only if the cutset also contains a CCF (such as shown for cutset #6).

- For cutset #6, a cutset coupling factor is calculated by multiplying the BE coupling factors for all BEs in the cutset. (Note that this calculation for cutset #6 is trivial since the calculation is: $0.2 \times 1.0 \times 1.0$.)

- As shown in cutsets #4 and #5, the CEM approach does not apply a coupling factor for cutsets that only contain HFEs that have cross-unit dependencies. This treatment is nonconservative but has no significant impact on MUCDF since the cutsets also contain several random BEs whose collective failure probability is very low. In addition, the failure probabilities for the dependent HFEs are relatively high, so the degree of nonconservatism is fairly low.

- As shown in cutset #7, coupling factors are not used for random failures and independent HFEs (MUCDF is obtained by multiplying the cutset CDF by only the U1 CCDP).

Table I-7 Significant CCF BEs, Human Failure Events, and Other BEs for LOOPWR Cutsets

U1 BE Probability	Name	Description	BE Coupling Factor	# OF OCCURRENCES IN 315 Cutsets	Coupling RULES (and MU CCF group sizes)
Significant CCF Basic Events					
3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN	0.2	1	4 of 4 RAT circuit breakers fail (all combinations)
2.15E-04	1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE	0.2	1	4 of 4 sequencers fail (all combinations)
3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN	0.2	5	4 of 4 DGNs fail (all combinations)
3.68E-05	1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START	0.2	3	4 of 4 DGNs fail (all combinations)
3.53E-05	1-EPS-MDP-FS-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO START	0.2	3	4 of 4 DGNs fail (all combinations)
1.19E-05	1-SWS-MOV-CF-1668A69A	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN	0.2	4	4 of 4 NSCW valves fail (all combinations)
8.70E-07	1-SWS-MOV-CF-116-ABCDEF	System Generated Event based upon Rasp CCF event : 1-SWS-MOV-CF-116	1	1	12 of 12 SWS MOVs fail
4.21E-06	1-SWS-MDP-CF-FS-ABCDEF	System Generated Event based upon Rasp CCF event : 1-SWS-MDP-CF-FS	1	2	12 of 12 SWS pumps fail
Operator Actions					
5.80E-02	1-OAB_TR-------H	OPERATORS FAIL TO FEED & BLEED - TRANSIENT	5.80E-02	13	No coupling between U1 and U2
5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO	5.73E-02	51	No coupling between U1 and U2
AFW-TDP CCF					
3.80E-02	1-AFW-TDP-FR-P4001___	TDAFWP (P4-001) FAILS TO RUN	0.2	22	2 of 2 TDPs fail
5.93E-03	1-AFW-TDP-FS-P4001___	TDAFWP (P4-001) FAILS TO START	0.2	4	2 of 2 TDPs fail
RANDOM - RCP					
2.00E-01	1-RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	0.2	19	No coupling between U1 and U2

Table I-7 Significant CCF BEs, Human Failure Events, and Other BEs for LOOPWR Cutsets (cont.)

U1 BE Probability	Name	Description	BE Coupling Factor	# OF OCCURRENCES IN 315 Cutsets	Coupling RULES (and MU CCF group sizes)
AC Power Recovery AND Convolution Factors					
5.59E-01	1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)	1	190	Failure of AC power recovery affect both units
3.64E-01	1-OEP-XHE-XX-NR02HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (2HR-WR AVAIL)	1	1	Failure of AC power recovery affect both units
4.86E-01	1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)	1	53	Failure of AC power recovery affect both units
6.87E-01	1-OEP-XHE-XL-NR01HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR (WEATHER-RELATED)	1	31	Failure of AC power recovery affect both units
3.13E-01	1-OEP-XHE-XX-NR01HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (1HR-WR AVAIL)	1	4	Failure of AC power recovery affect both units
Other CCF2 Combinations					
5 of 6 and 4 of 6 NSCW combinations (many combinations) appearing individually in cutsets			1		Assume full coupling between U1 and U2

Table I-8 Illustrative Cutset Review for Unit 1 LOOPWR

Cutset Type	Cutset #	Prob/Freq (/rcy)	Total % of CDF	Cutset	Description	BE Coupling Factor	Cutset Coupling Factor
	Total	8.56E-06	100	Displaying 315 Cut Sets. (14554 Original)	Top 95% contribution to total LOOPWR CDF of 9.015E-06/rcy		
CCF1	1	1.37E-06	15.98				0.2
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN	0.2	
RANDOM	2	8.64E-07	10.1				
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)		
		3.64E-01		1-OEP-XHE-XX-NR02HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (2HR-WR AVAIL)		
CCF1	3	8.40E-07	9.81				0.2
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE	0.2	
RANDOM	4	4.41E-07	5.15				
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE		
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)		
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)		

Table I-8 Illustrative Cutset Review for Unit 1 LOOPWR (cont.)

Cutset Type	Cutset #	Prob/Freq (/rcy)	Total % of CDF	Cutset	Description	BE Coupling Factor	Cutset Coupling Factor
RANDOM	5	4.41E-07	5.15				
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE		
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)		
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)		
CCF1	6	3.44E-07	4.02				0.2
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN	0.2	
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)	1	
		4.86E-01		1-OEP-XHE-XX-NR02HWR0	CONVOLUTION FACTOR FOR CCF-OPR (2HR-WR Avail)	1	
RANDOM-HEP	7	2.44E-07	2.85				
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER-RELATED)		
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)		
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO		

Table I-9 illustrates how the CEM approach was applied to certain selected Unit 1 LOOPWR cutsets. The following can be seen in Table I-9:

- The cutsets are organized by cutset type (and not by contribution to Unit 1 CDF or MUCDF contribution).

- The cutset coupling factor is either equivalent to the BE coupling factor or calculated using BE coupling factors. In turn, BE coupling factors have different assigned values depending on the cutset type, that is:

    - CCFs in cutsets type "CCF1" are typically assigned a BE coupling factor of 0.2. Similarly, CCFs in cutsets type "CCF2" are typically assigned a BE coupling factor of 1.0. If no other BEs appear in the cutset, then the BE coupling factor is the same as the cutset coupling factor.

    - If there are multiple BEs in a Unit 1 cutset, then a cutset coupling factor is calculated. For example, a cutset coupling factor has been calculated for cutset #27 (which is assigned the "CCF1+HEP" cutset type).

    - There is no coupling factor for cutsets that are assigned as a "RANDOM" cutset type.

- MUCDF contributions from each Unit 1 cutset are calculated in the following way:

    $MUCDF_i = CDF_i \times$ [Dependent & Independent Contributions] $\times$ [MUIEF / UIEF]

    where:

        Dependent contribution = Cutset coupling factor (CF)

        Independent contribution = Unit 1 CCDP

    And the rare events approximation is used:

        [Dependent & Independent] = CF + CCDP – (CF $\times$ CCDP)

- MUCDF based on the analyzed U1 cutsets is obtained by summing all MUCDF contributions shown in the far-right column in Table I-9.

- Total MUCDF for this MUIE is obtained by applying the scale-up factor (generally, 1.05, if U1 cutsets contributing 95 percent of U1 CDF are analyzed).

Table I-9 Illustration of MUCDF Calculations

Cutset Type	#	Cutset CDF Contribution (/rcy)	% of Total Unit 1 CDF	Cutset Coupling Factor	Unit 1 CCDP	Dependent & Independent	MUCDF (/rcy)
		CDF(i)		Cutset CF	CCDP		
CCF1	1	1.368E-06	15.98	0.2	2.31E-03	2.02E-01	1.726E-07
CCF1	3	8.399E-07	9.81	0.2	2.31E-03	2.02E-01	1.060E-07
CCF1	6	3.438E-07	4.02	0.2	2.31E-03	2.02E-01	4.337E-08
CCF1	25	8.031E-08	0.94	0.2	2.31E-03	2.02E-01	1.013E-08
CCF1	26	7.710E-08	0.9	0.2	2.31E-03	2.02E-01	9.726E-09
CCF1	43	2.345E-08	0.27	0.2	2.31E-03	2.02E-01	2.958E-09
CCF1	57	1.586E-08	0.19	0.2	2.31E-03	2.02E-01	2.001E-09
CCF1	102	6.121E-09	0.07	0.2	2.31E-03	2.02E-01	7.722E-10
CCF1	159	3.038E-09	0.04	0.2	2.31E-03	2.02E-01	3.833E-10
CCF1	168	2.657E-09	0.03	0.2	2.31E-03	2.02E-01	3.352E-10
CCF1+HEP	27	7.255E-08	0.85	1.15E-02	2.31E-03	1.37E-02	6.230E-10
CCF1+TDP-CCF	60	1.439E-08	0.17	0.04	2.31E-03	4.22E-02	3.797E-10
CCF2	67	9.202E-09	0.11	1	2.31E-03	1.00E+00	5.751E-09
CCF2	122	4.371E-09	0.05	1	2.31E-03	1.00E+00	2.732E-09
CCF2-RCP	158	3.293E-09	0.04	0.2	2.31E-03	2.02E-01	4.154E-10
RANDOM	2	8.644E-07	10.1		2.31E-03	2.31E-03	1.246E-09
RANDOM	4	4.410E-07	5.15		2.31E-03	2.31E-03	6.355E-10
RANDOM	5	4.410E-07	5.15		2.31E-03	2.31E-03	6.355E-10
RANDOM	8	1.873E-07	2.19		2.31E-03	2.31E-03	2.699E-10
RANDOM	9	1.873E-07	2.19		2.31E-03	2.31E-03	2.699E-10
RANDOM	10	1.473E-07	1.72		2.31E-03	2.31E-03	2.123E-10
RANDOM	12	1.166E-07	1.36		2.31E-03	2.31E-03	1.680E-10
RANDOM	14	1.119E-07	1.31		2.31E-03	2.31E-03	1.612E-10
RANDOM	15	1.029E-07	1.2		2.31E-03	2.31E-03	1.483E-10
RANDOM	17	9.520E-08	1.11		2.31E-03	2.31E-03	1.372E-10
RANDOM	21	9.169E-08	1.07		2.31E-03	2.31E-03	1.321E-10
RANDOM	23	8.095E-08	0.95		2.31E-03	2.31E-03	1.167E-10
RANDOM	28	6.966E-08	0.81		2.31E-03	2.31E-03	1.004E-10
RANDOM	30	5.690E-08	0.66		2.31E-03	2.31E-03	8.199E-11
RANDOM	32	4.336E-08	0.51		2.31E-03	2.31E-03	6.248E-11
RANDOM	35	3.542E-08	0.41		2.31E-03	2.31E-03	5.104E-11
RANDOM	38	3.437E-08	0.4		2.31E-03	2.31E-03	4.953E-11
RANDOM	40	3.084E-08	0.36		2.31E-03	2.31E-03	4.444E-11
RANDOM	45	2.191E-08	0.26		2.31E-03	2.31E-03	3.157E-11
RANDOM	56	1.748E-08	0.2		2.31E-03	2.31E-03	2.519E-11
RANDOM	61	1.400E-08	0.16		2.31E-03	2.31E-03	2.017E-11
RANDOM	62	1.117E-08	0.13		2.31E-03	2.31E-03	1.610E-11
RANDOM	68	8.787E-09	0.1		2.31E-03	2.31E-03	1.266E-11
RANDOM	78	7.525E-09	0.09		2.31E-03	2.31E-03	1.084E-11
RANDOM	93	7.245E-09	0.08		2.31E-03	2.31E-03	1.044E-11
RANDOM	107	5.914E-09	0.07		2.31E-03	2.31E-03	8.522E-12
RANDOM	111	5.509E-09	0.06		2.31E-03	2.31E-03	7.939E-12

I.2 CEM Approach Omissions in MUCDF Calculations

If each CCF modeled in the Unit 1 PRA appeared only once in cutset results, then the CEM calculations discussed above would be equivalent to PRA logic modeling of cross-unit CCFs. For the L3PRA project's Unit 1 PRA results, there are initiators for which a CCF appears only once in dominant cutsets. For example, two CCFs modeled in LOOPWRs (see example in the previous section) appear only once in the top 95 percent of cutsets. For all the significant CCFs in the LOOPWR cutsets:

- two CCFs appear only once
- two CCFs appear in three different cutsets
- one CCF appears in four cutsets
- one CCF appears in five cutsets

For the CCFs that appear in multiple cutsets, there are cross-combinations that would be generated in a fault tree-event tree PRA model. However, the CEM does not account for these cross-combinations. Sample calculations have indicated that MUCDF contributions from these "missing cutsets" can be small. Appendix J provides further discussion and example calculations of such omitted, CCF cross-combinations for MUCDF.

Similarly, Unit 1 and Unit 2 dependencies for operator actions have been addressed only if a CCF appears in the same cutset with the operator actions. Hand calculations presented in Appendix J for these dependencies also indicate that these contributions are small.

I.3 Results for MUCDF Calculations

This section summarizes the MUCDF results that have been developed for the ISR task using the CEM approach, referred to below as the base case MUCDF results. This section also provides results for a "FLEX sensitivity case" that addresses FLEX strategies and some other plant updates (e.g., new RCP seals).

I.3.1 Base Case MUCDF Results

Base case MUCDF calculations have been performed for all the initiating events that were identified as multi-unit initiating events (or sitewide initiating events). The results of these calculations are summarized in the sections below:

Section I.3.1.1 LOOPs

Section I.3.1.2 Internal fires

Section I.3.1.3 Seismic events

Section I.3.1.4 High winds

Section I.3.1.5 Loss of NSCW

An overall summary of the MUCDF results is provided in Section I.3.1.6.

I.3.1.1 Base Case MUCDF Results for LOOPs

Table I-10 shows the MUCDF results for all LOOPs. These results were developed by implementing the CEM approach with only the initial iteration of cutset review (see Section 6.2.3.2 of the main report for discussion of cutset review iterations). Results are given for:

- Grid-related LOOPs (LOOPGRs)
- Plant-centered LOOPs (LOOPPCs)
- Switchyard-centered LOOPs (LOOPSCs)
- Weather-related LOOPs (LOOPWRs)

As noted previously, additional iterations of review for cutsets with MU CCFs were performed for LOOPWRs. However, the calculated MUCDF for this more complete demonstration of the CEM approach was not much different than the result produced for the original (single cutset review) results (i.e., 4.35×10^{-7} per reactor-critical-year versus 4.47×10^{-7}per reactor-critical-year). In order to compare MUCDF results for LOOPs, only the original quantification results for LOOPWRs are shown in Table I-10.

Of particular interest are the highlighted rows of results that show:

- LOOPGRs make the largest contribution to MUCDF results (i.e., 55 percent).

- LOOPWRs and LOOPSCs make the next largest contributions to MUCDF (i.e., approximately 25 percent and 20 percent, respectively).

- LOOPPCs make very little contribution to MUCDF results (i.e., approximately 1 percent).

- The fraction of the Unit 1 CDF that represents MUCDF is 0.05 or less (i.e., 5 percent or less of the Unit 1 CDF) for all LOOP categories.

Table I-10 Summary Table for MUIEF and MUCDF for LOOP Events

	LOOPGR	LOOPPC	LOOPSC	LOOPWR		Total
U1IEF (/rcy)	1.23E-02	1.93E-03	1.04E-02	3.91E-03		2.85E-02
U1CDF (/rcy)	1.83E-05	1.91E-06	1.04E-05	9.02E-06		3.96E-05
U1-CCDP	1.49E-03	9.91E-04	9.95E-04	2.31E-03		1.39E-03
MUIEF (/rcy)	6.15E-03	1.07E-04	2.80E-03	2.44E-03		1.15E-02
MUCDF (/rcy)	1.00E-06	1.43E-08	3.57E-07	4.47E-07		1.82E-06
MU-CCDP	1.63E-04	1.33E-04	1.27E-04	1.83E-04		1.58E-04
% of total LOOP MUCDF	55.1%	0.8%	19.6%	24.6%		100.0%
MU Scenario Name	MU-IE-LOOPGR	MU-IE-LOOPPC	MU-IE-LOOPSC	MU-IE-LOOPWR		
Ratio MUCDF / U1CDF	0.05	0.01	0.03	0.05		0.05

I.3.1.2 Base Case MUCDF Results for Internal Fires

Table I-11 shows the MUCDF results for internal fires. For the MU-FIRE scenarios, the cutset method is not applicable. This is because the exact nature of many of the fire scenarios was not well understood for dual-unit assessment since the utility fire PRA model was adapted. Furthermore, thousands of fire sequences from the utility model were mapped into 210 different fire scenarios for the L3PRA project fire PRA, further masking a reliable assessment of MU potential. (Note that, for all hazards combined, the L3PRA project reactor, at-power, PRA models include 289 initiating events [i.e., scenarios/event trees]—internal fire scenarios accounted for 210.)

Consequently, internal fire estimates are performed at the fire scenario level while all other hazard category scenario estimates are performed at the cutset level. Four different fire scenarios were identified in the Phase 1 sitewide dependency assessment as being multi-unit scenarios. These four scenarios are:

MU-IE-FRI-1: Fires with conditional core damage probability equal to 1.0 (i.e., all main control room [MCR] abandonment scenarios and two yard fire scenarios)

MU-IE-FRI-2: Fires in shared areas

MU-IE-FRI-3: High CCDP scenarios originating in U1 and potentially affecting U2

MU-IE-FRI-4: High CCDP scenarios originating in U2 and potentially affecting U1

Note the following from the MUCDF results for fire:

- Almost half of the MUCDF is associated with main control room abandonment fires. It is assumed that both Unit 1 and Unit 2 control rooms are abandoned if there is a fire in either control room. MU-IE-FRI-1 scenarios are deemed conservative because no credit is given for use of the remote shutdown panels. A sensitivity analysis crediting plant operation from the remote shutdown panels was performed for the SUPRA, as documented in Section 19.4.3.2 of NRC (2023e). However, since the contribution of the MU-IE-FRI-1 scenarios to overall MU-CDF is not significant, no such sensitivity analysis was performed for the MUPRA. Note, the LERF effect may be more significant than the CDF effect.

- Fires in one unit that affect the other account for nearly 45 percent of the total MUCDF from internal fires. This contribution is also deemed conservative because, due to lack of a Unit 2 FPRA, it was assumed that if a fire originating in U1 results in CD in U1, then the CCDP for U2 is the same as that for U1. To limit the extent of this conservatism, the CCDP for U2 was capped at a value of 1.58×10^{-3}, which is the highest observed U1-CCDP for fires originating in U2.

- The results suggest that there would be one MU internal fire event every 20 years of reactor-critical operation, resulting in an MUCDF of 3×10^{-7}/rcy.

- The MUCDF is 0.5 percent of the Unit 1 internal fire events CDF (i.e., 6.14×10^{-5}/rcy) that was calculated in Level 3-PRA project's fire PRA. This is primarily because MCR abandonment scenarios are the only modeled fire scenarios that have a high degree of dependence between the two units.

- It was judged that an attempt to further break down the current four MU fire scenarios would introduce additional modeling assumptions but would not reduce the modeling uncertainty or provide a better estimate or new insights.

Table I-11 MUCDF for Four Internal Fire Scenarios

Scenario/ Parameter	Main Control Room*	Shared areas	U1 (High CCDP) to U2	U2 (High CCDP) to U1	Total
U1IEF (/rcy)	1.47E-07	3.42E-02	9.08E-03	**	
U1CDF (/rcy)	1.47E-07	1.00E-05	4.23E-05	**	
U1-CCDP	1.000	2.94E-04	4.66E-03	**	
MU Scenario Name	**MU-IE-FRI-1**	**MU-IE-FRI-2**	**MU-IE-FRI-3**	**MU-IE-FRI-4**	
MUIEF (/rcy)	1.47E-07	3.42E-02	9.08E-03	9.08E-03	5.23E-02
MUCDF (/rcy)	1.47E-07	2.28E-08	6.59E-08	6.59E-08	3.02E-07
MU-CCDP	1.0E+00	6.7E-07	7.3E-06	7.3E-06	5.77E-06
% total fire MUCDF	48.8%	7.5%	21.8%	21.8%	100.00%
Ratio MUCDF / U1CDF	1.0E+00	2.3E-03	1.6E-03		4.9E-03***

*Includes 12 MCR and 2 "YARD" fire sequences with CCDP = 1.
**Not used for estimation of MUCDF (IEF and CDF estimations for MU-IE-FRI-4 are modeled to be the same as MU-IE-FRI-3, by symmetry).
***Based on total CDF from the single unit fire PRA (6.14E-05/rcy).

I.3.1.3 Base Case MUCDF Results for Seismic Events

Table I-12 shows the MUCDF results for seismic events. These results were developed by implementing the CEM approach with only the initial iteration of cutset review. From Table I-12, the following can be observed:

- Seismic bins 5–8: The MUCDF is identical, or nearly identical, to the U1CDF due to the complete, or nearly complete, dependence between the units as a result of the MU seismic hazard correlations used in the MUCDF calculations.

- Seismic bins 3 and 4: The MUCDF is about 50 percent and 75 percent of the U1CDF for bins 3 and 4, respectively. For both bins, MU seismic hazard correlations dominate the MUCDF results, but bin 4 has more cutsets with MU dependencies than bin 3 does (i.e., 68 versus 59 SU cutsets with MUCDF contributions).

- Seismic bin 2: The MUCDF is about 10 percent of the U1CDF, and the largest contributing dependencies arise from cross-unit CCFs (though there are also some significant MUCDF contributions from cutsets that have MU seismic hazard correlations applied).

- Seismic bin 1: The MUCDF is 6 percent of the U1CDF, and the principal dependencies that drive these results are cross-unit CCFs (similar to LOOPs). These results imply that the effect of MU CCFs on MUCDF is smaller than the effect from MU seismic hazard correlations.

Table I-12 MUCDF for Seismic Events

	BIN-1	BIN-2	BIN-3	BIN-4	BIN-5	BIN-6	BIN-7	BIN-8	Total
pga =	**0.17g**	**0.39g**	**0.59g**	**0.79g**	**1.0g**	**1.29g**	**1.94g**	**2.5+g**	
U1IEF (/rcy)	1.64E-03	2.19E-04	4.79E-05	1.34E-05	4.26E-06	1.92E-06	2.48E-07	2.32E-09	1.93E-03
U1CDF (/rcy)	1.30E-06	1.22E-06	1.62E-06	2.43E-06	2.24E-06	1.75E-06	2.34E-07	2.32E-09	1.08E-05
U1-CCDP	0.001	0.01	0.03	0.18	0.53	0.91	0.94	1.00	5.60E-03
MU Scenario Name	**MU-IE-EQK-1**	**MU-IE-EQK-2**	**MU-IE-EQK-3**	**MU-IE-EQK-4**	**MU-IE-EQK-5**	**MU-IE-EQK-6**	**MU-IE-EQK-7**	**MU-IE-EQK-8**	
MUIEF (/rcy)	1.64E-03	2.19E-04	4.79E-05	1.34E-05	4.26E-06	1.92E-06	2.48E-07	2.32E-09	1.93E-03
MUCDF (/rcy)	2.07E-08	1.24E-07	8.20E-07	1.82E-06	2.06E-06	1.72E-06	2.34E-07	2.32E-09	6.80E-06
MU-CCDP	1.3E-05	5.6E-04	1.7E-02	0.14	0.48	0.90	0.94	1.00	3.53E-03
% total seismic MUCDF	0.3%	1.8%	12.1%	26.8%	30.3%	25.3%	3.4%	0.03%	100.00%
Ratio MUCDF / U1CDF	0.02	0.10	0.51	0.75	0.92	0.98	1.00	1.00	0.63

I.3.1.4 Base Case MUCDF Results for High Winds

Table I-13 shows the MUCDF results for high wind events. These results were developed by implementing the CEM approach with treatment of MU CCFs. Like other external hazards, the initiating event frequency used in the single unit model is actually a sitewide initiating event frequency already (i.e., the Unit 1 IE frequency and MUIE frequency are identical).

It should be noted that that there were 12 wind scenarios that were combined into 1 scenario for this calculation. For this reason, implementation of the CEM approach was a bit different than for LOOPs. For example:

- Six thousand (6000) cutsets were needed to represent 95 percent of the total SUCDF for all 12 wind scenarios. Cutsets for all 12 wind scenarios were combined into one Excel spreadsheet.

- For wind events, the only dependencies identified and addressed for the two reactors were MU CCFs. Both CCF types “CCF1” and “CCF2” were identified in cutset reviews.

- By using Excel’s “FIND” function, 745 of these 6000 cutsets were identified as containing CCFs. Although this number of CCFs is large compared to that for other MUIEs, it is still a relatively small number compared to the total number of SU cutsets for all wind events. Consequently, independent (or random failure) contributions to MUCDF were calculated first for all wind cutsets (both those with CCFs and those with random events only). Then, MU CCF contributions were calculated for the cutsets containing CCFs, using the appropriate CCF coupling factors. The independent and CCF contributions were added together to obtain the overall MUCDF results. (Note that this differs from the CEM implementation for LOOPs, where each cutset was handled individually with independent and dependent contributions calculated for each cutset.)

- Cutset reviewed identified that there were relatively few CCF2 type cutsets (i.e., cutsets containing certain NCSW failures). To take advantage of this fact, calculation of MUCDF contributions from CCFs was performed differently for wind events than for other MUIEs. Namely, MUCDF contributions were calculated for all cutsets that were **not** identified as type “CCF2.” Then, MUCDF contributions were calculated for the cutsets that contained relevant NSCW CCFs only. These two CCF contributions were added together for the overall CCF contribution to MUCDF.

Table I-14 shows the contributions from each of the 12 wind scenarios and the summations that were performed to develop the overall results shown in Table I-13. In addition to MUCDF calculations, Table I-14 also shows the calculations for MIN-MUCDF and MAX-MUCDF. Note that, because the MUIE frequency and Unit 1 IE frequency are identical, the maximum MUCDF is equivalent to the single unit CDF. Also, this means that MIN-MUCDF is equivalent to the independent (or random) contribution to overall MUCDF for wind events ($8.89{\times}10^{-8}$/rcy), which is added to the CCF contribution ($7.03{\times}10^{-7}$/rcy) to arrive at the total MU-WIND CDF ($7.93{\times}10^{-7}$/rcy).

Using the results shown in Table I-14, the fraction of MUCDF involving MU CCFs is 0.887.

Table I-13 MUCDF for High Wind Events (HWD+TOR)

	WIND (HWD+TOR)
U1IEF (/rcy)	8.89E-03
U1CDF (/rcy)	1.38E-05
U1-CCDP	0.0016
MUIEF (/rcy)	8.89E-03
MUCDF (/rcy)	7.93E-07
MU-CCDP	8.92E-05
MU Scenario Name	MU-IE-WIND-1
Ratio MUCDF / U1CDF	0.057

Table I-14 MUCDF Calculations for Individual Wind Scenarios and Overall Results (Non-FLEX Case)

Scenario #	Scenario Name	Scenario Description	MUIEF (=U1IEF) (/rcy)	CCDP	U1-CDF (= MAX-MUCDF) (/rcy)	MIN-MUCDF (/rcy)
			f	a	b	a*b
1	1-IE-HWD-BIN-1	STRAIGHT LINE WIND EVENT BIN 1 (95 MPH)	6.30E-03	5.25E-04	3.31E-06	1.74E-09
2	1-IE-HWD-BIN-2	STRAIGHT LINE WIND EVENT BIN 2 (110 MPH)	1.16E-03	4.00E-03	4.63E-06	1.85E-08
3	1-IE-HWD-BIN-3	STRAIGHT LINE WIND EVENT BIN 3 (129 MPH)	1.37E-04	1.57E-02	2.15E-06	3.37E-08
4	1-IE-HWD-BIN-4	STRAIGHT LINE WIND EVENT BIN 4 (156 MPH)	7.02E-06	3.45E-02	2.42E-07	8.36E-09
5	1-IE-TOR-BIN-WM-1	TORNADO EVENT BIN 1 (85 MPH) - Plant Area	4.60E-04	3.66E-04	1.68E-07	6.14E-11
6	1-IE-TOR-BIN-WP-1	TORNADO EVENT BIN 1 (85 MPH) - Point Target	1.06E-04	6.95E-04	7.37E-08	5.12E-11
7	1-IE-TOR-BIN-WM-2	TORNADO EVENT BIN 2 (110 MPH) - Plant Area	3.69E-04	1.95E-03	7.19E-07	1.40E-09
8	1-IE-TOR-BIN-WP-2	TORNADO EVENT BIN 2 (110 MPH) - Point Target	4.55E-05	8.68E-03	3.95E-07	3.43E-09
9	1-IE-TOR-BIN-WM-3	TORNADO EVENT BIN 3 (135 MPH) - Plant Area	1.72E-04	4.11E-03	7.06E-07	2.90E-09
10	1-IE-TOR-BIN-WP-3	TORNADO EVENT BIN 3 (135 MPH) - Point Target	2.19E-05	1.96E-02	4.29E-07	8.40E-09
11	1-IE-TOR-BIN-WM-4	TORNADO EVENT BIN 4 (165 MPH) - Plant Area	9.91E-05	7.04E-03	6.98E-07	4.92E-09
12	1-IE-TOR-BIN-WP-4	TORNADO EVENT BIN 4 (165 MPH) - Point Target	1.21E-05	2.31E-02	2.79E-07	6.45E-09
	WIND = HWD + TOR	Summed results (RANDOM or independent)	8.89E-03		1.38E-05	8.99E-08
		CCF contribution (CCF1 + CCF2).				7.03E-07
	WIND (TOTAL)	MU-WIND CDF (CCF + Random) (TOTAL)				7.93E-07

I.3.1.5 Base Case MUCDF Results for Loss of NSCW

Table I-15 shows the MUCDF results for loss of NSCW events. These results were developed by implementing the CEM approach with only the initial iteration of cutset review.

The analysis for loss of NSCW contains many assumptions, starting with the assumption that the initiating event frequency for a single unit is the same as for both units (i.e., complete dependence). Similar assumptions regarding dependencies between certain NSCW components (e.g., pumps) for the two units are made in the MUCDF calculations. (See Appendix H for further information on coupling factors for NSCW.) These dependencies explain the results shown in Table I-15, such as the MUCDF for losses of NCSW being approximately 37 percent of the single unit CDF.

Table I-15 MUCDF for LONSCW Events

	MU-LONSCW
U1IEF (/rcy)	3.47E-05
U1CDF (/rcy)	8.76E-06
U1-CCDP	0.252
MUIEF (/rcy)	3.47E-05
MUCDF (/rcy)	3.23E-06
MU-CCDP	9.31E-02
MU Scenario Name	MU-IE-LONSCW
Ratio MUCDF / U1CDF	0.37

I.3.1.6 Summary of Results

Table I-16 shows all the base case MUCDF results. From Table I-16, it is seen that:

- Seismic events contribute over half of total MUCDF (approximately 55 percent), almost entirely coming from bins 3 through 6 (approximately 52 percent of total MUCDF). Further, bins 4 through 6 contribute approximately 45 percent of total MUCDF, in nearly equal shares.

- The contribution of losses of NCSW is about 26 percent of the total MUCDF contribution.

- LOOP events collectively contribute about 15 percent of total MUCDF, with grid-related LOOPs contributing the most (approximately 8 percent of the total).

- Wind-related events and internal fires are minor contributors to MUCDF (contributing approximately 2 percent each).

Since the estimated contribution to MUCDF from the loss of NSCW initiating event is significant and perhaps unexpected, some background on how MUCDF was developed for this MUIE is provided. In particular, two modeling assumptions used to estimate MUCDF for loss of NSCW contribute to this result:

- The LONSCW initiating event frequency was modeled in the L3PRA by a fault tree. This FT is dominated by CCFs of NSCW pumps, which show up as BEs in the CDF cutsets. The estimation of MUCDF assumed that, if this initiating event occurs, it affects both units; namely, these BEs were assigned a BE coupling factor of 1.0. This assumption is consistent with what is also assumed for other MU events (e.g., cutset type CCF2) for those failures (but not as initiating events). If this assumption is considered too pessimistic, a reduction in the coupling factor could be used to reduce the estimated MUCDF by the same ratio. For example, if it is assumed that only 50 percent of the Unit 1 LONSCW initiating events affect Unit 2 as an initiating event, then the estimated MUCDF would likewise be reduced by 50 percent.

- A considerable number of significant Unit 1 PRA LONSCW cutsets include HFEs. Accordingly, these cutsets were assigned the "HEP" cutset type, with a BE coupling factor of 0.1. This BE coupling factor assignment implies high correlation between the HFEs in both units. If this assumption is considered as unduly pessimistic, a reduction in the coupling factor could reduce the estimated MUCDF by up to 20 percent.

Table I-16 MUCDF Estimates – Base Case

	Scenario Name	Scenario Description	MU Scenario Characteristics	MUIEF (/rcy)	MUCDF (/rcy)	% MUCDF
1	MU-IE-LOOPGR	Grid-Related LOOP	SBO and AC power recovery failure	6.15E-03	1.00E-06	8.1%
2	MU-IE-LOOPPC	Plant-Centered LOOP	SBO and AC power recovery failure	1.07E-04	1.43E-08	0.1%
3	MU-IE-LOOPSC	Switchyard-Centered LOOP	SBO and AC power recovery failure	2.80E-03	3.56E-07	2.9%
4	MU-IE-LOOPWR	Weather-Related LOOP	SBO and AC power recovery failure	2.44E-03	4.47E-07	3.6%
5	MU-LONSCW	Loss of NSCW	Loss of NSCW in both units	3.47E-05	3.23E-06	25.9%
6	MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned with CCDP =1	1.47E-07	1.47E-07	1.2%
7	MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2	at least MU LOOP (assumed)	3.42E-02	2.28E-08	0.18%
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.53%
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.53%
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1–0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic BIN-1	1.64E-03	8.08E-08	0.65%
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3–0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic BIN-2	2.19E-04	1.24E-07	1.0%
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5–0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic BIN-3	4.79E-05	8.26E-07	6.6%
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7–0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic BIN-4	1.34E-05	1.84E-06	14.8%
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9–1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic BIN-5	4.26E-06	2.02E-06	16.3%
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1–1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic BIN-6	1.92E-06	1.72E-06	13.9%
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5–2.5g) occurs (bin pga 1.94g)	2-unit SBO and Major structural damage (EQK-BIN7) with CCDP =1	2.48E-07	2.34E-07	1.9%
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and Major structural damage (EQK-BIN8) with CCDP = 1	2.32E-09	2.32E-09	0.02%
18	MU-IE-WIND-1	SBO and SSC wind damage	SBO and WIND damage to SSCs	8.89E-03	2.32E-07	1.9%
			Total =	7.47E-02	1.24E-05	100.0%

I.3.2 FLEX Sensitivity Case MUCDF Results

FLEX sensitivity case MUCDF calculations have been performed for LOOPs, seismic events, and losses of NSCW. The FLEX sensitivity case includes credit for declaration of extended loss of AC power (ELAP) and implementation of FLEX strategies,[88] and new RCP shutdown seals. Otherwise, the same assumptions are used as for the base case.

Table I-17 shows MUCDF results for both the base case (for all MUIEs) and FLEX case (for selected MUIEs). Along with the base case and FLEX sensitivity case MUCDF results, "FLEX effectiveness" results for MUCDF also are shown (as was done for the single unit PRAs). Similar to the single unit CDF results, the MUCDF results show the following regarding FLEX effectiveness:

- The biggest impact on MUCDF results is for all LOOPs and losses of NSCW.
    - LOOPGR – 86%
    - LOOPPC – 90%
    - LOOPSC – 87%
    - LOOPWR – 87%
    - LONSCW – 94%
- There are significant impacts on MUCDF results for seismic bins 1 through 5
    - Bin 1 – 30%
    - Bin 2 – 24%
    - Bin 3 – 29%
    - Bin 4 – 27%
    - Bin 5 – 16%
- There is little impact on the MUCDF results for seismic bins 6, 7, and 8.

Table I-18 provides different types of FLEX MUCDF results for seismic events only. First, FLEX MUCDF results are compared with respect to contributions from the different dependencies represented in the MUCDF results (e.g., structural failures, cross-unit CCFs, human failure events). For example:

- Structural failures contribute to all (or almost all) of the MUCDF for seismic bins 7 and 8.
- Structural failures make no (or almost no) contribution to MUCDF for seismic bins 1 and 2.
- MU CCFs make no (or almost no) contribution to MUCDF for seismic bins 4 through 8.
- MU CCFs make some contribution to the MUCDF for seismic bins 1 through 3.

[88] 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.

- Cross-unit human dependencies make no contribution to seismic bins 4 through 8.
- Cross-unit human dependencies make very little contribution to seismic bins 1 through 3.

Then, for seismic bin 2 MUCDF results only, Table I-18 shows the contributions from these same dependencies for both the base and FLEX sensitivity cases. For all MU dependencies explored, the percent contribution is very similar (e.g., in both cases, the “seismic” contribution is around 60 percent and the cross-unit CCF contribution is around 30 percent).

Table I-17 MUCDF Estimates – FLEX and No-FLEX

				NO-FLEX Case				WITH-FLEX Case		FLEX Effectiveness	Comment	NO-FLEX CCDP
	Scenario Name	**Scenario Description**	**MU Scenario Characteristics**	**MU-IEF**	**MU-CDF**	**% MU-CDF**		**MU-CDF**		**(a-b)/a**		**c/f**
				f	**a**	**c**		**b**				
1	MU-IE-LOOPGR	Grid-Related LOOP	SBO and AC power recovery failure	6.15E-03	1.00E-06	7.7%		1.45E-07		85.5%		1.63E-04
2	MU-IE-LOOPPC	Plant-Centered LOOP	SBO and AC power recovery failure	1.07E-04	1.43E-08	0.1%		1.42E-09		90.1%		1.33E-04
3	MU-IE-LOOPSC	Switchyard-Centered LOOP	SBO and AC power recovery failure	2.80E-03	3.57E-07	2.7%		4.81E-08		86.5%		1.27E-04
4	MU-IE-LOOPWR	Weather-Related LOOP	SBO and AC power recovery failure	2.44E-03	4.47E-07	3.4%		5.65E-08		87.4%		1.83E-04
5	MU-LONSCW	Loss of NSCW	Loss of NSCW in both units	3.47E-05	3.23E-06	24.8%		1.87E-07		94.2%		9.30E-02
6	MU-IE-FRI-1	MCR abandonment due to fire	Both MCRs are abandoned with CCDP =1	1.47E-07	1.47E-07	1.1%		1.47E-07		0.0%		1.00E+00
7	MU-IE-FRI-2	Shared (A+Y) area fires by U1 and U2	at least MU LOOP (assumed)	3.42E-02	2.28E-08	0.18%						6.67E-07
8	MU-IE-FRI-3	U1 to U2 (U1 fires affecting U2)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.51%						7.26E-06
9	MU-IE-FRI-4	U2 to U1 (U2 fires affecting U1)	at least (other unit reactor trip and fire damage) (assumed)	9.08E-03	6.59E-08	0.51%						7.26E-06
10	MU-IE-EQK-1	Seismic event in bin 1 (0.1 - 0.3g) occurs (bin pga 0.17g)	2-unit SBO due to CCFs in seismic BIN-1	1.64E-03	8.08E-08	0.62%		5.60E-08		30.6%		4.93E-05
11	MU-IE-EQK-2	Seismic event in bin 2 (0.3 - 0.5g) occurs (bin pga 0.39g)	2-unit SBO due to CCFs in seismic BIN-2	2.19E-04	1.24E-07	1.0%		9.35E-08		24.3%		5.64E-04
12	MU-IE-EQK-3	Seismic event in bin 3 (0.5 - 0.7g) occurs (bin pga 0.59g)	2-unit SBO and seismic SSC damage in seismic BIN-3	4.79E-05	8.26E-07	6.4%		5.91E-07		28.5%	?	1.72E-02
13	MU-IE-EQK-4	Seismic event in bin 4 (0.7 - 0.9g) occurs (bin pga 0.79g)	2-unit SBO and seismic SSC damage in seismic BIN-4	1.34E-05	1.84E-06	14.2%		1.36E-06		26.5%	?	1.38E-01
14	MU-IE-EQK-5	Seismic event in bin 5 LOOP (0.9 - 1.1g) occurs (bin pga 1.0g)	2-unit SBO and seismic SSC damage in seismic BIN-5	4.26E-06	2.02E-06	15.6%		1.70E-06		15.8%		4.75E-01
15	MU-IE-EQK-6	Seismic event in bin 6 LOOP (1.1 - 1.5g) occurs (bin pga 1.29g)	2-unit SBO and seismic SSC damage in seismic BIN-6	1.92E-06	1.72E-06	13.3%		1.68E-06		2.6%		8.97E-01
16	MU-IE-EQK-7	Seismic event in bin 7 LOOP (1.5 - 2.5g) occurs (bin pga 1.94g)	2-unit SBO and Major structural damage (EQK-BIN7) with CCDP =1	2.48E-07	2.34E-07	1.8%		2.34E-07		0.0%		9.43E-01
17	MU-IE-EQK-8	Seismic event in bin 8 LOOP (2.5g and above) occurs (bin pga 2.5g)	2-unit SBO and Major structural damage (EQK-BIN8) with CCDP = 1	2.32E-09	2.32E-09	0.02%		2.32E-09		0.0%		1.00E+00
18	MU-IE-WIND-1	SBO and SSC wind damage	SBO and WIND damage to SSCs	8.89E-03	7.93E-07	6.1%		2.32E-07		70.7%		8.92E-05
			Total =	7.47E-02	1.30E-05	100.0%		6.53E-06		49.7%		1.74E-04

Note: All IEF and CDF values are in terms of per reactor-critical-year (rcy).

Table I-18 Observations for the MU FLEX Seismic Scenarios

	BIN-1	BIN-2	BIN-3	BIN-4	BIN-5	BIN-6	BIN-7	BIN-8
pga =	**0.17g**	**0.39g**	**0.59g**	**0.79g**	**1.0g**	**1.29g**	**1.94g**	**2.5+g**
CDF due to "STRUCTURE" failures (direct CD) (/rcy)	0.00E+00	0.00E+00	8.08E-08	2.74E-07	3.76E-07	7.21E-07	2.34E-07	2.32E-09
BIN CCDP for structural failures	0.00E+00	0.00E+00	0.002	0.02	0.09	0.38	0.94	1.00
"STRUCTURE" contribution	none	ignorable	some	some	some	some	almost ALL	ALL
"SEISMIC" contribution	NO	YES	YES	YES	YES	YES		
"CCF" contribution	some	some	some	very little	none	none	none	none
"HEP" contribution	almost none	almost none	almost none	almost none	none	none	none	none
Seismically-induced LOOP probability	0.13	0.70	0.92	0.98	1.00	1.00	1.00	1.00
BIN-2 Contributors (only)		**WITH-FLEX**	**NO-FLEX**					
	Seismic	64%	60%					
	CCF	29%	31%					
	Others	2%	5%					
	Structure	5%	4%					
	Total	100%	100%					
	Total MUCDF	9.35E-08	1.24E-07					
	FLEX Effectiveness	24%						

APPENDIX J
TESTING AND SEPARATE CALCULATIONS FOR CUTSET ESTIMATION METHOD

This appendix provides a summary of alternate calculations done to “test” the results of the cutset estimation method (CEM) approach that is described in Section 6 and Appendix I.

J.1 Introduction

The L3PRA project team recognizes that the CEM approach is an estimation of multi-unit core damage frequency (MUCDF). Consequently, efforts were made to understand the differences between the CEM approach and the traditional PRA modeling approach. Section J.2 compares the two approaches using a simple example. Section J.3 documents the results of hand calculations performed to verify that some simplifications (omissions) in the CEM approach do not result in significant underestimation of MUCDF.

J.2 Simple Comparison Between Fault Tree Linking and CEM Approach

A simple comparison of the CEM approach and the fault tree linking approach implemented in traditional single unit PRAs was performed. This simple example, consisting of a switchyard-centered loss of offsite power (LOOP) and only three cutsets, is presented here to illustrate the CEM approach and compare the CEM approach with the fault tree (FT) linking method.

The CEM approach described in this report is used because: (1) a Unit 1 core damage frequency (CDF) model exists, and a Unit 2 CDF model does not, and (2) both units on the reference site are essentially identical. Thus, an examination of Unit 1 CDF cutsets, classification of them according to their potential coupling with Unit 2 for multi-unit (MU) scenarios already defined, and assigning coupling factors to them can be used to estimate the two-unit (i.e., MU) CDF, scenario by scenario.

In this example, MUCDF results were developed as shown in Table J-1 (for the CEM approach) and Table J-2 (for the FT-linking approach using SAPHIRE). Table J-1 shows CEM results similar to those provided in Appendix I, such as the cutsets that were selected, the color-coding of cutsets by cutset type, the assignment of BE coupling factors, and the calculation of MUCDF (both individual cutset contributions and total).

In turn, Table J-2 shows the results of applying the FT-linking method in SAPHIRE, starting with the original three Unit 1 cutsets and ending up with nine MU cutsets. The same MU initiating event (MUIE) frequency and basic event (BE) coupling factors were used as for the CEM approach.

The two tables show that the MUCDF results produced by each method are identical, that is:

- MUCDF (CEM): 2.06×10^{-7}/reactor-critical-year (rcy)
- MUCDF (FT linking): 2.06×10^{-7}/rcy

However, it should be noted that the fault tree linking approach will always produce a greater level of completeness for more complex cases.

Table J-1 Illustration of CEM

U1-IE Frequency =		1.04E-02	a				
U1CDF =		1.11E-06	b				
U1 CCDP =		1.07E-04	c - b/a				
MUIE Frequency =		2.80E-03	d	Substituted below replacing U1-IE			
				Cutset frequency (e) is re-calculated with it.			
Cutset Type	**#**	**Cutset Frequency**			**Coupling Factor**	**Cutset U2-CCDP**	**Cutset MUCDF**
		e			**f**	**g=f+c-f*c**	**h=e*f**
CCF-A	1	9.79E-07				2.00E-01	1.96E-07
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED)			
		3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U1 CRBs TO OPEN	0.2		
CCF-B	2	5.20E-08				2.00E-01	1.04E-08
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))			
		3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF U1 DGNS 1 AND 2 TO RUN	0.2		
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE U1 SYSTEMS AFTER AC RECOVERED IN SBO			
RANDOM	3	8.01E-08				1.07E-04	8.55E-12
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))			
		5.35E-03	1-ACP-CRB-CC-AA0205__	U1 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN			
		5.35E-03	1-ACP-CRB-CC-BA0301__	U1 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN			
(Total is the MUCDF before scaling-up.)				Total =			**2.06E-07**

Table J-2 Illustration of FT Linking

U1 and U2 CDF Cutsets are AND-gated in SAPHIRE (With symmetry and "full 2U coupling" modeling assumptions)				
Models both units undergoing a MU-LOOPSC event.				
First, 9 cutsets for MU LOOPSC event are generated with total independence assumption. Then coupling factors are substituted by SAPHIRE post-processing rules.				
Orange	marks "2U coupling" factors (0.20 and 1.0) substituted into cutsets.			
MU CDF =		2.06E-07	Sum of frequencies of 9 cutsets below. Compare with CEM results in Table 6-3.	
Cutset Type	**CS #**	**Freq/Prob**	**BE Name**	**BE Description**
CCF-A-CCF-A	1	1.96E-07		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U1 CRBs TO OPEN
		2.00E-01	2-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U2 CRBs TO OPEN - conditional
CCF-A-CCF-B	2	1.82E-11		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U1 CRBs TO OPEN
		3.24E-04	2-EPS-DGN-CF-FRUN1	CCF OF U2 DGNS 3 AND 4 TO RUN
		5.73E-02	2-OA-ORS-------H	OPERATORS FAIL TO RESTORE U2 SYSTEMS AFTER AC RECOVERED IN SBO
CCF-A-RANDOM	3	2.80E-11		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U1 CRBs TO OPEN
		5.35E-03	2-ACP-CRB-CC-AA0205__	U2 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	2-ACP-CRB-CC-BA0301__	U2 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

Table J-2 Illustration of FT Linking (cont.)

CCF-B-CCF-A	4	1.82E-11		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF U1 DGNS 1 AND 2 TO RUN
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE U1 SYSTEMS AFTER AC RECOVERED IN SBO
		3.50E-04	2-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U2 CRBs TO OPEN
CCF-B-CCF-B	5	1.04E-08		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF U1 DGNS 1 AND 2 TO RUN
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE U1 SYSTEMS AFTER AC RECOVERED IN SBO
		2.00E-01	2-EPS-DGN-CF-FRUN1	CCF OF U2 DGNS 3 AND 4 TO RUN (given U1 EDGs failed by common cause.)
		1.00E+00	2-OA-ORS-------H	OPERATORS FAIL TO RESTORE U2 SYSTEMS AFTER AC RECOVERED IN SBO
CCF-B-RANDOM	6	1.49E-12		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF U1 DGNS 1 AND 2 TO RUN
		5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE U1 SYSTEMS AFTER AC RECOVERED IN SBO
		5.35E-03	2-ACP-CRB-CC-AA0205__	U2 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	2-ACP-CRB-CC-BA0301__	U2 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
RANDOM-CCF-A	7	2.80E-11		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		5.35E-03	1-ACP-CRB-CC-AA0205__	U1 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	1-ACP-CRB-CC-BA0301__	U1 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.50E-04	2-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC U2 CRBs TO OPEN

Table J-2   Illustration of FT Linking (cont.)

RANDOM-CCF-B	8	1.49E-12		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		5.35E-03	1-ACP-CRB-CC-AA0205__	U1 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	1-ACP-CRB-CC-BA0301__	U1 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.24E-04	2-EPS-DGN-CF-FRUN1	CCF OF U2 DGNS 3 AND 4 TO RUN
		5.73E-02	2-OA-ORS-------H	OPERATORS FAIL TO RESTORE U2 SYSTEMS AFTER AC RECOVERED IN SBO
RANDOM-RANDOM	9	2.29E-12		
		2.80E-03	12-IE-LOOPSC	MU LOOP (SWITCHYARD-CENTERED))
		5.35E-03	1-ACP-CRB-CC-AA0205__	U1 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	1-ACP-CRB-CC-BA0301__	U1 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	2-ACP-CRB-CC-AA0205__	U2 RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03	2-ACP-CRB-CC-BA0301__	U2 RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

J.3 Hand Calculations of Cross-Combinations of Dependent Events Omitted from CEM Approach

This section documents the results of hand calculations performed for omitted MU cutsets, including "cross-combinations" of common-cause failures (CCFs) in MU cutsets that are omitted from the CEM approach. The purpose of these hand calculations is to verify that the omission of such cross-combinations in the CEM approach does not result in significant underestimation of MUCDF. For the purposes of these calculations, the MU weather-related LOOP (LOOPWR) results are used.

J.3.1 Problem Statement

As stated in Section I.2, if each CCF modeled in the Unit 1 PRA appeared only once in cutset results, then the CEM calculations discussed above would be equivalent to PRA logic modeling of cross-unit CCFs. For the L3PRA project's Unit 1 PRA results, there are initiators for which a CCF appears only once in dominant cutsets. For example, two CCFs modeled in LOOPWR appear only once in the top 95 percent of cutsets (see example in Section I.1.4). For all the significant CCFs in the LOOPWR cutsets:

- two CCFs appear only once
- two CCFs appear in 3 different cutsets
- one CCF appears in 4 cutsets
- one CCF appears in 5 cutsets

For the CCFs that appear in multiple cutsets, there are cross-combinations that would be generated in a fault tree-event tree PRA model. However, the CEM does not necessarily account for all these cross-combinations. The CEM quantification approach multiplies each Unit 1 CCF cutset with both a BE (or cutset) coupling factor and the total Unit 2 conditional core damage probability (CCDP) for the MUIE being analyzed. While multiplying by the total Unit 2 CCDP accounts for all the cutset cross-combinations, this approach implicitly assumes that none of the other Unit 2 cutsets share any dependencies with the Unit 1 CCF cutset. If additional dependencies exist between the cutsets, the CEM approach may lead to underestimation of MUCDF.

Sample calculations have indicated that MUCDF contributions from omitting these additional dependencies can be small. The following subsections provide further discussion and example calculations related to these omissions.

Also, for many of the CEM calculations performed for the ISR task, dependencies between Unit 1 and Unit 2 operator failures were addressed only if they appeared in a cutset with a CCF. In these CEM calculations, the BE coupling factor for the operator failure is implicitly taken as 1.0 if the cutset also contains a CCF (i.e., the cutset probability was calculated without explicitly using a coupling factor or a random failure probability so the default value is 1). Cutsets containing dependent operator failures and no CCF were not addressed in most CEM calculations.

In the hand calculations presented in Section J.3.3, both cross-unit (or MU) CCFs and operator failure dependencies are addressed.

J.3.2 How Many Cutsets are Important?

For the purposes of this discussion, Table I-7 is duplicated below as Table J-3. The first information from Table J-3 that is important is the number of occurrences of CCFs (and human failures) in the 315 Unit 1 cutsets used for applying the CEM approach in the LOOPWR sensitivity case. For example:

- The first two CCFs listed occur only once each. Consequently, there are no cross-combinations to address.
- CCF of emergency diesel generators (EDGs) to run occurs five times.
- CCF of EDGs to start occurs three times.
- CCF of EDG fuel transfer pumps to start occurs three times.
- CCF of nuclear service cooling water (NSCW) containment spray valves HV1668A and HV1669A to open occurs four times.
- The system-generated CCF event for NSCW motor-operated valve (MOV) failures (1-SWS-MOV-CF-116) occurs only once so there are no cross-combinations to consider for this CCF.
- The system-generated CCF event for NSCW pump failures (1-SWS-MDP-CF-FS) occurs twice.
- Turbine-driven auxiliary feedwater (TDAFW) pump (P4-001) fails to run occurs 22 times.
- TDAFW pump (P4-001) fails to start occurs four times.

The number of CCF occurrences is not high, but the number of potential cross-combinations could be.

Table J-3 Significant CCF BEs, Human Failure Events, and Other BEs for LOOPWR Cutsets

U1 BE Probability	Name	Description	BE Coupling Factor	Number of Occurrences in 315 Cutsets	Coupling Rules (and MU CCF Group Sizes)
Significant CCF Basic Events					
3.50E-04	1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN	0.2	1	4 of 4 RAT circuit breakers fail (all combinations)
2.15E-04	1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE	0.2	1	4 of 4 sequencers fail (all combinations)
3.24E-04	1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN	0.2	5	4 of 4 DGNs fail (all combinations)
3.68E-05	1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START	0.2	3	4 of 4 DGNs fail (all combinations)
3.53E-05	1-EPS-MDP-FS-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO START	0.2	3	4 of 4 DGNs fail (all combinations)
1.19E-05	1-SWS-MOV-CF-1668A69A	CCF OF NSCW CT SPRAY VALVES HV1668A & 1669A TO OPEN	0.2	4	4 of 4 NSCW valves fail (all combinations)
8.70E-07	1-SWS-MOV-CF-116-ABCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MOV-CF-116	1	1	12 of 12 SWS MOVs fail
4.21E-06	1-SWS-MDP-CF-FS-ABCDEF	System Generated Event based upon Rasp CCF event: 1-SWS-MDP-CF-FS	1	2	12 of 12 SWS pumps fail
Operator Actions					
5.80E-02	1-OAB_TR-------H	OPERATORS FAIL TO FEED & BLEED - TRANSIENT	5.80E-02	13	No coupling between U1 and U2
5.73E-02	1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO	5.73E-02	51	No coupling between U1 and U2
AFW-TDP CCF					
3.80E-02	1-AFW-TDP-FR-P4001___	TDAFWP (P4-001) FAILS TO RUN	0.2	22	2 of 2 TDPs fail
5.93E-03	1-AFW-TDP-FS-P4001___	TDAFWP (P4-001) FAILS TO START	0.2	4	2 of 2 TDPs fail
RANDOM - RCP					
2.00E-01	1-RCS-MDP-LK-BP2	RCP SEAL STAGE 2 INTEGRITY (BINDING/POPPING OPEN) FAILS	0.2	19	No coupling between U1 and U2

Table J-3 Significant CCF BEs, Human Failure Events, and Other BEs for LOOPWR Cutsets (cont.)

U1 BE Probability	Name	Description	BE Coupling Factor	Number of Occurrences in 315 Cutsets	Coupling Rules (and MU CCF Group Sizes)
AC Power Recovery AND Convolution Factors					
5.59E-01	1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)	1	190	Failure of AC power recovery affect both units
3.64E-01	1-OEP-XHE-XX-NR02HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (2HR-WR AVAIL)	1	1	Failure of AC power recovery affect both units
4.86E-01	1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)	1	53	Failure of AC power recovery affect both units
6.87E-01	1-OEP-XHE-XL-NR01HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 1 HOUR (WEATHER-RELATED)	1	31	Failure of AC power recovery affect both units
3.13E-01	1-OEP-XHE-XX-NR01HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (1HR-WR AVAIL)	1	4	Failure of AC power recovery affect both units
Other CCF2 Combinations					
5 of 6 and 4 of 6 NSCW combinations (many combinations) appearing individually in cutsets			1		Assume full coupling between U1 and U2

Table J-3 shows many more occurrences of operator actions (and associated convolution factors). For example, the human failure event “Operators fail to recover offsite power in 2 hours (weather-related)” occurs 190 times.

Table J-4 shows the top 30 cutsets for LOOPWR. This table shows that:

- By the 30th cutset, the percent contribution to the overall CDF is less than 1 percent (i.e., 0.66 percent).

- Many of the cutsets in the top 30 are assigned a cutset type of “RANDOM” (i.e., there are no cross-unit dependencies to address).

- Some of the operator actions are independent (i.e., they have no cross-unit dependencies)

Going further down the cutset list obviously shows even smaller contributions to total CDF (e.g., the 50th cutset contributes 0.23 percent, the 60th cutset contributes 0.17 percent, the 70th cutset contributes 0.1 percent). Such contributions suggest that it is not necessary to do hand calculations for many cutset combinations. The hand calculations below, which apply to both CCFs and operator actions (both HFEs and convolution factor BEs), use this conclusion to limit the number of hand calculations performed.

Table J-4 shows that operator actions that have identified cross-unit dependencies, and are ***not*** in a cutset with a CCF, are in cutsets that have multiple random failures and have been assigned a RANDOM cutset type.

For all calculations, the following inputs are needed:

- Unit 1 IE frequency (U1IEF) = 3.91×10^{-3}/rcy
- Unit 1 CDF (U1CDF) = 9.02×10^{-6}/rcy
- Unit 1 CCDP (U1-CCDP)* = 2.31×10^{-3}
- MUIE frequency (MUIEF) = 2.44×10^{-3}/rcy
- MUCDF = 4.13×10^{-7}/rcy (without scale-up)

(*assumed to be equal to the Unit 2 CCDP)

Coupling factors for the relevant CCFs also were used in these calculations.

Table J-4 Top 30 Cutsets for LOOPWR

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
	Total	8.56E-06	100	Displaying 315 CutSets. (14554 Original)	Top 95% contribution to total LOOPWR CDF of 9.015E-06
CCF1	1	1.37E-06	15.98		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.50E-04		1-ACP-CRB-CF-A205301	CCF OF SWITCHYARD AC CRBs AA205 & BA301 TO OPEN
RANDOM	2	8.64E-07	10.1		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		3.64E-01		1-OEP-XHE-XX-NR02HWR2	CONVOLUTION FACTOR FOR 2FTR-OPR (2HR-WR AVAIL)
CCF1	3	8.40E-07	9.81		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		2.15E-04		1-EPS-SEQ-CF-FOAB	CCF OF SEQUENCERS TO OPERATE

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
RANDOM	4	4.41E-07	5.15		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	5	4.41E-07	5.15		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
CCF1	6	3.44E-07	4.02		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR0	CONVOLUTION FACTOR FOR CCF-OPR (2HR-WR Avail)
RANDOM-HEP	7	2.44E-07	2.85		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
RANDOM	8	1.87E-07	2.19		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	9	1.87E-07	2.19		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	10	1.47E-07	1.72		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
RANDOM	11	1.47E-07	1.72		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
RANDOM	12	1.17E-07	1.36		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	13	1.17E-07	1.36		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	14	1.12E-07	1.31		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
RANDOM	15	1.03E-07	1.2		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		2.94E-03		1-EPS-DGN-FS-G4001___	DG1A RANDOMLY FAILS TO START
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	16	1.03E-07	1.2		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		2.94E-03		1-EPS-DGN-FS-G4002___	DG1B RANDOMLY FAILS TO START
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	17	9.52E-08	1.11		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		2.72E-03		1-DCP-BAT-MA-BD1B____	BATTERY 1BD1B IN MAINTENANCE
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM	18	9.52E-08	1.11		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		2.72E-03		1-DCP-BAT-MA-AD1B____	BATTERY 1AD1B IN MAINTENANCE
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
		4.86E-01		1-OEP-XHE-XX-NR02HWR1	CONVOLUTION FACTOR FOR 1FTR-OPR (2HR-WR AVAIL)
RANDOM+HEP	19	9.31E-08	1.09		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		3.30E-02		1-EPS-DGN-FR-G4001___	DG1A RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
RANDOM+HEP	20	9.31E-08	1.09		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.30E-02		1-EPS-DGN-FR-G4002___	DG1B RANDOMLY FAILS TO RUN (24 HR MISSION TIME)
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO
RANDOM	21	9.17E-08	1.07		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
RANDOM	22	9.17E-08	1.07		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
RANDOM	23	8.10E-08	0.95		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		2.94E-03		1-EPS-DGN-FS-G4001___	DG1A RANDOMLY FAILS TO START
		1.26E-02		1-EPS-DGN-MA-G4002___	DG1B IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
RANDOM	24	8.10E-08	0.95		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		2.94E-03		1-EPS-DGN-FS-G4002___	DG1B RANDOMLY FAILS TO START

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
		1.26E-02		1-EPS-DGN-MA-G4001___	DG1A IN MAINTENANCE
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
CCF1	25	8.03E-08	0.94		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.68E-05		1-EPS-DGN-CF-FSUN1	CCF OF UNIT 1 DGNs G4001/G4002 TO START
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
CCF1	26	7.71E-08	0.9		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.53E-05		1-EPS-MDP-FS-XFERPPS_-CC	CCF OF DG FUEL TRANSFER PUMPS TO START
		5.59E-01		1-OEP-XHE-XL-NR02HWR	OPERATORS FAIL TO RECOVER OFFSITE POWER IN 2 HOURS (WEATHER-RELATED)
CCF1+HEP	27	7.26E-08	0.85		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		3.24E-04		1-EPS-DGN-CF-FRUN1	CCF OF UNIT 1 DGNS G4001/G4002 TO RUN
		5.73E-02		1-OA-ORS-------H	OPERATORS FAIL TO RESTORE SYSTEMS AFTER AC RECOVERED IN SBO

Table J-4 Top 30 Cutsets for LOOPWR (cont.)

Cutset Type	#	Prob/Freq	Total %	Cutset	Description
RANDOM	28	6.97E-08	0.81		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-AA0205__	RAT A SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.33E-03		1-EPS-SEQ-FO-1821U302	SEQUENCER B FAILS TO OPERATE
RANDOM	29	6.97E-08	0.81		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		3.33E-03		1-EPS-SEQ-FO-1821U301	SEQUENCER A FAILS TO OPERATE
RANDOM	30	5.69E-08	0.66		
		3.91E-03		1-IE-LOOPWR	LOSS OF OFFSITE POWER (WEATHER- RELATED)
		5.35E-03		1-ACP-CRB-CC-BA0301__	RAT B SUPPLY CRB RANDOMLY FAILS TO OPEN
		2.72E-03		1-DCP-BAT-MA-AD1B____	BATTERY 1AD1B IN MAINTENANCE

J.3.3 Hand Calculations for Cross-Combinations of Multi-Unit CCFs

An illustrative hand calculation of “cross-combinations” of one CCF was performed using the LOOPWR results. CCF of EDGs to run was selected because, of all CCFs that appear in more than one cutset, it appears both in the highest contributing cutset (i.e., cutset #6) and in the most number of cutsets (5) in the 315 cutsets that make up the top 95 percent of LOOPWR CDF. Specifically, this CCF appears in cutsets #6, #27, #60, #201, and #208.

EDG fuel transfer pumps fail to start is the CCF in the next highest cutset to contain a CCF. However, this is cutset #26 and the only other occurrences of this CCF are in cutsets #57 and #75.

The CEM approach already addresses the “mirror” combinations of CCF cutsets (i.e., each Unit 1 CCF cutset with its Unit 2 counterpart):

- Unit 1 cutset #6 and Unit 2 cutset #6
- Unit 1 cutset #27 and Unit 2 cutset #27
- Unit 1 cutset #60 and Unit 2 cutset #60
- Unit 1 cutset #201 and Unit 2 cutset #201
- Unit 1 cutset #208 and Unit 2 cutset #208

Based on the rationale presented in Section J.3.2, hand calculations were performed for the following additional cutset combinations:

- Unit 1 cutset # 6 and Unit 2 cutset #27
- Unit 1 cutset #6 and Unit 2 cutset #60
- Unit 1 cutset #27 and Unit 2 cutset #60

Because Units 1 and 2 are essentially identical, these combinations also address the combinations with “Unit 1” replaced with “Unit 2.” These calculations are given below.

The BE coupling factor for this CCF is 0.2.

J.3.3.1 Cross-Combination MUCDF Contribution: Cutset #6 and Cutset #27

Cutset #6 consists of the following BEs (in addition to the U1IEF) with a Unit 1 CDF contribution of 3.44×10^{-7}/rcy:

- CCF of EDGs to run
- Operators fail to recover offsite power in 2 hours (weather-related)
- Convolution factor for CCF-OPR (2 hr WR avail)

Cutset #27 consists of the following BEs (in addition to the U1EIF) and their associated failure probabilities and BE coupling factors:

- CCF of EDGs to run: 3.24×10^{-4}; BE coupling factor: 0.2
- Operators fail to restore system after AC recovery: 5.73×10^{-2} (independent failure)

The cutset coupling probability for cutset #27 is:

$$0.2 \times 5.73 \times 10^{-2} = 1.15 \times 10^{-2}$$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #6 CDF] × [Cutset #27 coupling probability]

= $(2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 3.44\times10^{-7}/\text{rcy} \times 1.15\times10^{-2}$

= $2.47\times10^{-9}/\text{rcy}$

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 4.94×10^{-9}/rcy. This value is about 1 percent of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor).[89]

J.3.3.2 Cross-Combination MUCDF Contribution: Cutset #6 and Cutset #60

Cutset #6 consists of the following BEs (in addition to the U1IEF) with a contribution of 3.44×10^{-7}/rcy to Unit 1 CDF:

- CCF of EDGs to run
- Operators fail to recover offsite power in 2 hours (weather-related)
- Convolution factor for CCF-OPR (2 hr WR avail)

Cutset #60 consists of the following BEs (in addition to the U1IEF) and their associated failure probabilities and BE coupling factors:

- CCF of EDGs to run: 3.24×10^{-4}; BE coupling factor: 0.2
- TDAFW pump fails to run: 3.8×10^{-2} / (BE coupling factor not relevant for this calculation)
- Operators fail to recover offsite power in 1 hour (weather-related): 6.87×10^{-1}; BE coupling factor: 1.0
- Convolution factor for CCF-OPR (1 hr WR avail): 4.35×10^{-1}; BE coupling factor: 1.0

The cutset coupling probability for cutset #60 is:

$0.2 \times 3.8\times10^{-2} \times 1.0 \times 1.0 = 7.6\times10^{-3}$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #6 CDF] × [Cutset #60 coupling probability]

= $(2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 3.44\times10^{-7}/\text{rcy} \times 7.6\times10^{-3}$

= $1.63\times10^{-9}/\text{rcy}$

[89] The underestimation of MUCDF from this omission is actually less than 4.94×10^{-9}/rcy since the contribution from cutset #6 for Unit 1 and the independent CCDP for Unit 2 is accounted for in the base quantification. This same caveat applies to the other omissions evaluated in this section.

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 3.26×10^{-9}/rcy. This value is less than 1 percent of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor).

J.3.3.3 Cross-Combination MUCDF Contribution: Cutset #27 and Cutset #60

Cutset #27 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 7.26×10^{-8}/rcy:

- CCF of EDGs to run
- Operators fail to restore system after AC recovery

Cutset #60 consists of the following BEs (in addition to the U1IEF) and their associated failure probabilities and BE coupling factors:

- CCF of EDGs to run: 3.24×10^{-4}; BE coupling factor: 0.2
- TDAFW pump fails to run: 3.8×10^{-2} / (BE coupling factor not relevant for this calculation)
- Operators fail to recover offsite power in 1 hour (weather-related): 6.87×10^{-1}; BE coupling factor: 1.0
- Convolution factor for CCF-OPR (1 hr WR avail): 4.35×10^{-1}; BE coupling factor: 1.0

The cutset coupling probability for cutset #60 is:

$$0.2 \times 3.8\times10^{-2} \times 1.0 \times 1.0 = 7.6\times10^{-3}$$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #27 CDF] × [Cutset #60 coupling probability]

$$= (2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 7.26\times10^{-8}/\text{rcy} \times 7.6\times10^{-3}$$

$$= 3.44\times10^{-10}/\text{rcy}$$

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 6.89×10^{-10}/rcy. This value is less than 1 percent (~0.2 percent) of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor).

As noted in Section J.3.2, combinations involving lower ranked cutsets would have even smaller MUCDF contributions.

J.3.4 Hand Calculations for Combinations of Multi-Unit Human Failure Events

An illustrative hand calculation of missing cross-unit dependencies for operator actions was performed using the LOOPWR results. From Table J-3 and Table J-4, it can be seen that the highest combination only appears once in cutsets (i.e., cutset #2):

- Operators fail to recover offsite power in 2 hours (weather-related)
- Convolution factor for 2FTR-OPR (2hr-WR avail) (this BE appears only once in LOOPWR cutsets)

For illustrative purposes, the combination of operator failure and convolution factor was chosen based on which BE appeared in the highest ranked cutsets. Again, from Table J-4, the BE chosen for hand calculations is:

- Convolution factor for 1FTR-OPR (2hr-WR avail)

According to Table J-3, there are 53 occurrences of this BE. Using Table J-4 and the Excel spreadsheets for the LOOPR calculations, the following cutsets were chosen for hand calculations: #4, #8, and #9. All these cutsets were assigned a cutset type of RANDOM.

The BE coupling factor for all operator actions for these BEs is 1.0.

J.3.4.1 Cross-Unit Operator Dependencies: Cutsets #4 and #8

Cutset #4 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 4.41×10^{-7}/rcy:

- DG1B randomly fails to run (24-hr mission time): 3.30×10^{-2}
- DG1A in maintenance: 1.26×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related): 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

Cutset #8 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 1.87×10^{-7}/rcy:

- RAT B supply circuit breaker randomly fails to open: 5.35×10^{-3}
- DG1A randomly fails to run (24-hr mission time): 3.30×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related) 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

The cutset coupling probability for cutset #8 (calculated from random failure probabilities and BE coupling factors are used) is:

$$(5.35\times10^{-3}) \times (3.30\times10^{-2}) \times 1.0 \times 1.0 = 1.77\times10^{-4}$$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #4 CDF] × [Cutset #8 coupling probability]

$$= (2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 4.41\times10^{-7}/\text{rcy} \times 1.77\times10^{-4}$$

$$= 4.86\times10^{-11}/\text{rcy}$$

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 9.72×10^{-11}/rcy. This value is approximately ~0.02 percent (i.e., considerably less than 1 percent) of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor).

J.3.4.2 Cross-Unit Operator Dependencies: Cutsets #4 and #9

Cutset #4 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 4.41×10^{-7}/rcy:

- DG1B randomly fails to run (24-hr mission time): 3.30×10^{-2}
- DG1A in maintenance: 1.26×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related): 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

Cutset #9 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 1.87×10^{-7}/rcy:

- RAT A supply circuit breaker randomly fails to open: 5.35×10^{-3}
- DG1B randomly fails to run (24-hr mission time): 3.30×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related): 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

The cutset coupling probability for cutset #9 (calculated from random failure probabilities and BE coupling factors are used) is:

$$(5.35\times10^{-3}) \times (3.30\times10^{-2}) \times 1.0 \times 1.0 = 1.77\times10^{-4}$$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #4 CDF] × [Cutset #9 coupling probability]

$$= (2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 4.41\times10^{-7}/\text{rcy} \times 1.77\times10^{-4}$$

$$= 4.86\times10^{-11}/\text{rcy}$$

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 9.72×10^{-11}/rcy. This value is approximately ~0.02 percent (i.e., considerably less than 1 percent) of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor). Note that this contribution is identical to that for the combination of cutsets #4 and #8 because the only difference between the cutsets is which train (A or B) has RAT or EDG failures.

J.3.4.3 Cross-Unit Operator Dependencies: Cutsets #8 and #9

Cutset #8 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 1.87×10^{-7}/rcy:

- RAT B supply circuit breaker randomly fails to open: 5.35×10^{-3}
- DG1A randomly fails to run (24-hr mission time): 3.30×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related) 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

Cutset #9 consists of the following BEs (in addition to the U1IEF) with a contribution to Unit 1 CDF of 1.87×10^{-7}/rcy:

- RAT A supply circuit breaker randomly fails to open: 5.35×10^{-3}
- DG1B randomly fails to run (24-hr mission time): 3.30×10^{-2}
- Operators fail to recover offsite power in 2 hours (weather-related): 5.59×10^{-1}
- Convolution factor for 1FTR-OPR (2hr-WR avail): 4.86×10^{-1}

The cutset coupling probability for cutset #9 (calculated from random failure probabilities and BE coupling factors are used) is:

$$(5.35\times10^{-3}) \times (3.30\times10^{-2}) \times 1.0 \times 1.0 = 1.77\times10^{-4}$$

So, the MUCDF contribution from this combination is:

MUIEF/U1IEF × [Cutset #8 CDF] × [Cutset #9 coupling probability]

$$= (2.44\times10^{-3}/\text{rcy} / 3.91\times10^{-3}/\text{rcy}) \times 1.87\times10^{-7}/\text{rcy} \times 1.77\times10^{-4}$$

$$= 2.07\times10^{-11}/\text{rcy}$$

Since there are two possible combinations of these cutsets (i.e., reversing Units 1 and 2), this value is multiplied by 2 to obtain a combined MUCDF contribution of 4.14×10^{-11}/rcy. This value is approximately ~0.01 percent (i.e., considerably less than 1 percent) of the total MUCDF (4.13×10^{-7}/rcy) that was calculated for LOOPWRs (before applying the scale-up factor).

As noted in Section J.3.2, contributions from lower ranked CDF cutsets would produce even smaller MUCDF contributions.

APPENDIX K

ESTIMATE MULTI-UNIT LEVEL 2 RISK

This appendix documents supporting information for the multi-unit (MU) Level 2 PRA work described in Section 8. Section K.1 documents the results of a sensitivity analysis investigating the likelihood of hydrogen combustion leading to failure of the auxiliary building. Section K.2 and Section K.3 address the quantification of MU release category frequencies (MURCFs). Section K.2 documents an approach that assumes the release categories for each unit are independent of each other. Section K.3 documents an approach that attempts to address inter-unit dependencies for the MU release categories.

K.1 MELCOR Sensitivity Analysis

This section provides information that is slightly modified from that given in a section of an older version of the Level 2 PRA for internal event and flooding phenomenological appendix. This information is included here because it is not part of any other public reports.

In this sensitivity analysis, the likelihood of hydrogen combustion outside containment and the impact of combustion on the integrity of the auxiliary building are investigated. The integrity of the auxiliary building has an influence on the releases to the environment in the event of containment bypass scenarios. Moreover, the auxiliary building pressure capacity of approximately 1.1 bar-abs is much lower than the containment. As a result, any global deflagration in the auxiliary building is assumed to lead to its failure.

Two particular scenarios are examined in detail for features such as time-dependent gas concentrations and combustion history in the auxiliary building:

- S5: This scenario is an interfacing systems loss-of-coolant accident (ISLOCA) initiated by a break to the auxiliary building in the residual heat removal (RHR) piping. One end of the broken RHR piping is connected to one of the hot legs thereby causing the break to discharge the reactor coolant system (RCS) coolant directly into the auxiliary building. The break is located in the next-to-lowest level (Level C) of the auxiliary building. The break elevation, with respect to the bottom of the compartment it discharges into, is specified such that the break is considered to be submerged. In addition, about 101 m^3 of liquid fills up the compartment into which the break discharges and results in a spillover of any additional liquid into the lower most level (with a large volume). Note that the actual details of the location and details of break (compartment, elevation of break, spill over volume etc.) are not definitively known due to lack of such information. In this scenario, the emergency core cooling system (ECCS) and refueling water storage tank (RWST) function normally, in that they are not directly affected by the break, and the liquid pumped by the six pumps reaches the cold legs as designed.

- S7: This scenario represents a station blackout event with the loss of containment isolation at the time of event initiation. The flow path representing the loss of containment isolation is located about 0.3 m from the basemat in the containment and connects to the third from grade level (Level C) of the auxiliary building. Reactor cooling pump (RCP) seal leaks begin at the start of the transient with nominal rate of 21 gpm per pump; the leakage rate (i.e., the leak flow area) is maintained constant throughout the accident. Rapid depressurization of the steam generators begins at 30 minutes via the atmospheric relief valves (ARVs). Auxiliary feedwater (AFW) is assumed to be unavailable, and all ARVs are closed after 4 hours.

In case S5, due to the discharge of hot liquid from the break and the resulting generation of steam through flashing, the steam concentration in the level is very high causing the environment to be steam inerted. Therefore, even though the hydrogen concentration reaches approximately 10 percent, no combustion is likely. This behavior is also noted for the lowest level of the auxiliary building. A decrease in the steam and hydrogen concentrations after approximately 12 hours is due to the occurrence of vessel breach (predicted at 11.8 hours).

An additional contributor to the substantially decreased likelihood for combustion in the auxiliary building in this scenario [ISLOCA] is found to be the ventilation system. The ventilation system removes substantial amounts of hydrogen thereby preventing the corresponding concentration from increasing to combustible concentrations anywhere in the auxiliary building including, levels that are not steam inerted. It is worth noting that the routing of the ductwork in the auxiliary building is not definitively known. In addition, neither has the likelihood of combustion in ventilation ducts been assessed nor is the likelihood of failure or clogging of the filters considered.

The hydrogen, oxygen, and steam concentrations in the level of the auxiliary building where the containment bypass is located for case S7 are shown in Figure K-1. The concentrations reach combustible levels and MELCOR does predict several deflagrations to occur. MELCOR also predicts the auxiliary building pressure to rise above 1.1 bar-abs during the first deflagration thereby leading to failure of the auxiliary building. The importance of the ventilation system is demonstrated once again because the unavailability of the system (due to station blackout) allows hydrogen concentrations to build up to combustible levels.

Based on the analysis of the MELCOR results for cases S5 and S7, it is concluded that from the standpoint of global deflagration, the survival of the auxiliary building is dependent on the continued operation of the ventilation fans.

Newer MELCOR results do not provide any additional information that would challenge the assumption that a combustion is likely in the absence of the ventilation system. That said, the prior value of 1.0 does not reflect any uncertainty in this situation, and so a value of 0.9 will be assigned moving forward. Meanwhile, the new results suggest that combustion is more likely than previously assessed in cases where the ventilation system is operating (given that the modeled cases narrowly avoided combustion). For this reason, a subjectively assigned value of 0.5 is assigned for the probability that a combustion will fail the auxiliary building when the ventilation system is operating. No calculations using ERPRA-BURN were performed to further refine these likelihoods, in light of other large uncertainties.

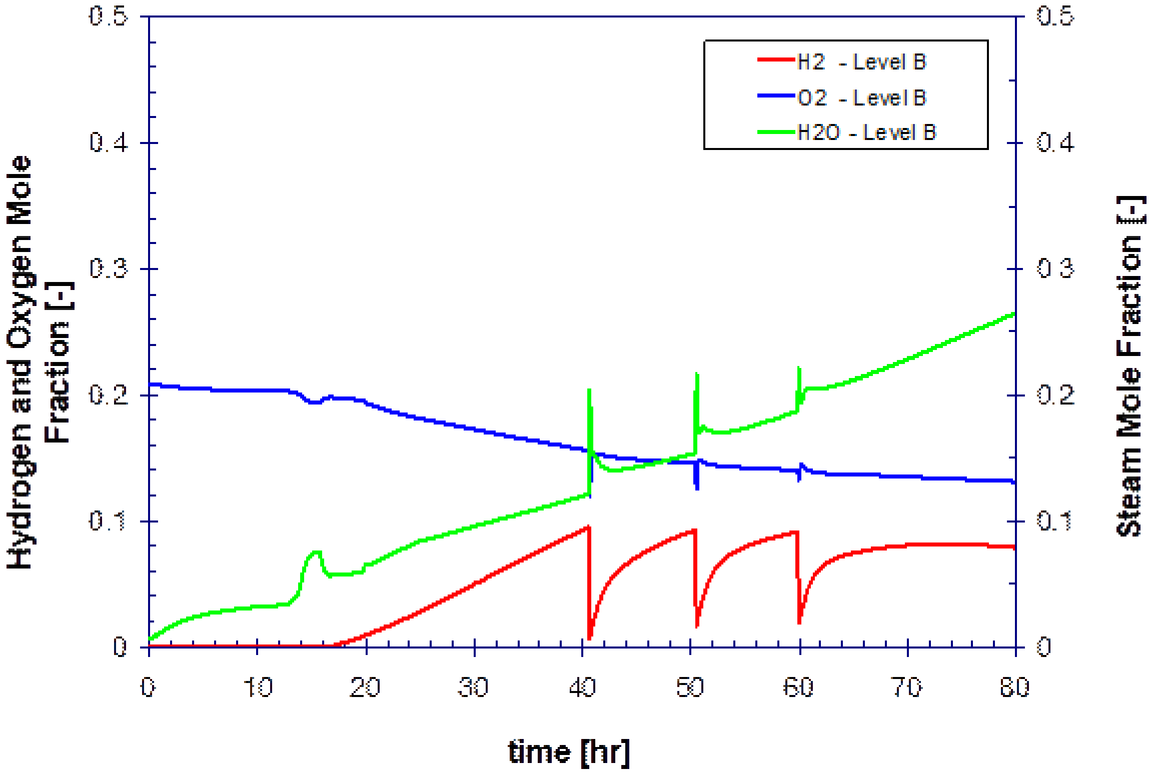

Figure K-1 Hydrogen and Steam Concentration in the Auxiliary Building Level that Communicates with the Containment due to Isolation Failure (Case S7)

K.2 Independent Process for Quantifying Multi-Unit Release Category Frequency

If the release categories for each unit are assumed to be independent, then the frequency of each two-unit release category combination can be estimated by multiplying the probabilities of the two release categories conditional on core damage. At each unit, the probability of a given release category is the fraction of single unit release frequency that is in that release category.

For example, in the case of the LOOPWR initiator, the single unit release frequencies are shown in Table K-1. The highest-frequency release category, 1-REL-LCF, makes up 46.4 percent of all releases. Assuming this is true at both Unit 1 and Unit 2, with no dependence between them, the probability of the LCF-LCF release category combination (conditional on multi-unit core damage) is 0.464 × 0.464 = 0.216. To get the overall frequency of this combination, this conditional probability is multiplied by the MU core damage frequency (MUCDF) (which is 4.35×10^{-7}/reactor-critical-year [rcy] for LOOPWR). This process is shown in Table K-2 and Table K-3 for the three release categories that make up 96 percent of LOOPWR release frequency.

Table K-1 LOOPWR Single Unit Release Category Frequencies

Name	Point Estimate (/rcy)	% of total
1-REL-BMT	1.65E-08	0.2%
1-REL-CIF	1.17E-08	0.1%
1-REL-CIF-SC	0.00E+00	0.0%
1-REL-ECF	1.34E-09	0.0%
1-REL-ICF-BURN	1.29E-06	12.0%
1-REL-ICF-BURN-SC	1.23E-07	1.1%
1-REL-ISGTR	1.07E-07	1.0%
1-REL-LCF	5.00E-06	46.4%
1-REL-LCF-SC	1.94E-07	1.8%
1-REL-NOCF	4.02E-06	37.4%
1-REL-SGTR-C	0.00E+00	0.0%
1-REL-SGTR-O	5.20E-11	0.0%
1-REL-SGTR-O-SC	5.56E-10	0.0%
1-REL-V	0.00E+00	0.0%
1-REL-V-F	0.00E+00	0.0%
1-REL-V-F-SC	0.00E+00	0.0%
Total	1.08E-05	100.0%

Table K-2 Independent Conditional Probability of LOOPWR Release Category Combinations

		Unit 1 Release Category		
		ICF-BURN	LCF	NOCF
Unit 2 Release Category	**Single Unit %**	12%	46%	37%
ICF-BURN	12%	0.014	0.055	0.045
LCF	46%	0.055	0.216	0.174
NOCF	37%	0.045	0.174	0.140
Sum of all combinations:		0.917 (91.7% of MUCDF)		

Table K-3 Independent Frequency of LOOPWR Release Category Combinations

	Unit 1 Release Category		
Unit 2 Release Category	**ICF-BURN**	**LCF**	**NOCF**
ICF-BURN	6.21E-09 /rcy	2.41E-08 /rcy	1.94E-08 /rcy
LCF	2.41E-08 /rcy	9.38E-08 /rcy	7.55E-08 /rcy
NOCF	1.94E-08 /rcy	7.55E-08 /rcy	6.08E-08 /rcy

For combinations in which the release categories at the two units are different, the total frequency of that combination should combine the two possible orderings (e.g., Unit 1 LCF and Unit 2 NOCF, plus Unit 1 NOCF and Unit 2 LCF). Therefore, the frequency of the LCF-NOCF release category combination is 7.55×10^{-8}/rcy $\times$ 2 = 1.51×10^{-7}/rcy.

K.3 Dependent Process for Quantifying Multi-Unit Release Category Frequency

The process for quantifying MURCF by the fault tree method involves the following steps:

1. Unlink any event trees that have been linked for the Level 2 PRA, except 1-FPI-LOOPWR (or whichever initiator is being quantified), and link and solve that one, with linkage rules set up to calculate RCFs (Level 2 analysis). Using multiple initiators at once is extremely inefficient.

   If the entire model has already been solved for RCFs, it should also be possible to achieve the same result by slicing the original end state cutsets by the initiating event instead of solving the event tree again.

2. Gather the relevant end states (those that have substantial frequency).

3. View the cutsets for each end state separately and use SAPHIRE's "Slice" feature to get a reasonable number of cutsets (preferably covering at least 95 percent of frequency). Save them to a new end state. Its name should include the unit (1-), the initiator (e.g. EQK6), the release category, and the percentage of frequency included.

4. Turn the cutsets into a fault tree using a new feature in SAPHIRE 8.2.8 or later. Right click on the end state and select Convert Cutsets to Fault Tree Logic, then Save As.

5. Right click on the new fault tree and choose Clone (Save As). Find "1-*" and replace it with "3-*" (where 3- is the prefix used to refer to Unit 2, to avoid confusion with some existing events that begin with 2-). Click Replace Events, then Replace Gates, then Save As. To speed up the calculation, you can also find "3-IE-LOOPWR" (or whatever the name of the initiating event is after replacing 1-* with 3-*) and replace it with 1-IE-LOOPWR, undoing the change to that one event. This will cause the combined cutsets to have one less basic event. Because every cutset will contain the same initiating event for both units, the post-processing rules can insert the multi-unit version of the initiator without checking for the Unit 2 initiating event.

6. Clone the fault tree again, this time changing 1-* to M-*, to create the M- events for multi-unit failures (it will also create some others that are not needed, and the new fault tree will not be needed either, but they can be safely ignored).

7. Determine which basic events used in these cutsets have a cross-unit dependency. These are found typically by looking at CCF events in the top 100 cutsets. Choose a coupling factor for each of them (the ratio of multi-unit failure to unit 1 failure of the same event).

8. Create as many different coupling factor basic events as you need. In this example, all coupling factors were rounded to the nearest 0.1 to limit the number of coupling factor basic events.

   These coupling factors should be named according to their probability; in this case, COUPLING-FACTOR-02 has probability 0.2, COUPLING-FACTOR-09 has probability 0.9, and COUPLING-FACTOR-01 has probability 0.1. For completely dependent events (coupling factor 1.0) no coupling factor basic event is needed.

9. Generate post-processing rules from the list of cross-unit CCFs, either using a script, or, if the number of coupled events is manageable, then by writing them manually based on the examples below.

   The purpose of this step is to create a rule for each dependent basic event that picks out cutsets where that event occurs at both units and replaces them with a multi-unit version of that event, as well as the appropriate coupling factor to adjust its probability. For example,

   ```
   if 1-ACP-CRB-CF-A205301 * 3-ACP-CRB-CF-A205301 then
      DeleteEvent = 1-ACP-CRB-CF-A205301;
      DeleteEvent = 3-ACP-CRB-CF-A205301;
      AddEvent = M-ACP-CRB-CF-A205301;
      AddEvent = COUPLING-FACTOR-02;
   endif
   ```

   The added coupling factor is only necessary if it is less than 1.0 (not completely dependent). If the model being solved includes success probabilities for some of the events with cross-unit dependencies, then those success events should also be correlated:

   ```
   if /1-ACP-CRB-CF-A205301 * /3-ACP-CRB-CF-A205301 then
      DeleteEvent = /1-ACP-CRB-CF-A205301;
      DeleteEvent = /3-ACP-CRB-CF-A205301;
      AddEvent = /M-ACP-CRB-CF-A205301;
      AddEvent = COUPLING-FACTOR-09;
   endif
   ```

   The coupling factor for the success probabilities is described in Section 8.2.2.3.

   For high probability basic events, such as those that have their success probabilities retained, it is also important to remove frequency from cutsets that are precluded by the cross-unit dependence. Where the dependence is total, it should not be possible for an event to succeed at one unit and fail at the other. For example:

   ```
   if /1-RCS-SLOCA-EQ6 * ~/3-RCS-SLOCA-EQ6 then
   ```

```
    DeleteRoot;
endif
if ~/1-RCS-SLOCA-EQ6 * /3-RCS-SLOCA-EQ6 then
    DeleteRoot;
Endif
```

In principle, this would also apply to fully dependent failures without success probabilities, so cutsets that contain the event for one unit and not the other should be removed. The difficulty with that approach would be identifying which cutsets queried the event in both units, since many would involve unrelated failure modes at the two units. However, if the basic event probability is small, all those cutsets that should be removed are low frequency and will mostly fall below the truncation limit anyway.

For the initiating event, rather than adding a coupling factor, it is more efficient to modify the probability of the multi-unit basic event (M-IE-LOOPWR in this case) to reflect the frequency of the multi-unit initiator.

```
if 1-IE-LOOPWR then
    DeleteEvent = 1-IE-LOOPWR;
    AddEvent = M-IE-LOOPWR;
Endif
```

10. Create an MU-* fault tree, named for the desired release category combination (e.g. MU-LCF-NOCF), that has a single AND gate, and under that gate put the 1-* and 3-* fault trees.

11. Add the rules generated above to the fault tree postprocessing rules for the MU- fault tree.

12. Right click on the fault tree and select Solve.

13. Set the truncation limit to a low probability. The postprocessing rules will increase the frequencies of cutsets with dependent failures, but before the rules are applied, the multi-unit cutsets will all have very low frequency and will be truncated prematurely if the limit is too high. If the number of cutsets in each of the RC fault trees is small (less than about 1000), the limit can be set so low that it captures all cutsets (e.g. 1e-20). If there are more cutsets, it may be necessary to start with a higher truncation limit ($\sim 10^{-14}$) and solve the tree several times with a lower limit each time. Continue until the result has converged, meaning that an order of magnitude decrease in the truncation limit does not increase the result by more than 5 percent.

14. Make sure the following options are checked: Solve for Cutsets, Apply Post-Processing Rules, Update / Quantify Cutsets, and Quantify Cutsets.

15. Solve the fault tree.

16. If necessary due to excessive frequency inflation, quantify the resulting cutsets using an alternate method as described in Section 8.3.3.

NRC FORM 335 (12-2010) NRCMD 3.7 — U.S. NUCLEAR REGULATORY COMMISSION **BIBLIOGRAPHIC DATA SHEET** *(See instructions on the reverse)*	1. REPORT NUMBER (Assigned by NRC, Add Vol., Supp., Rev., and Addendum Numbers, if any.) **NUREG-XXXX**
2. TITLE AND SUBTITLE **U.S. NRC Level 3 Probabilistic Risk Assessment Project** Integrated Site Risk, All Hazards, Level 1, Level 2, and Level 3 PRA	3. DATE REPORT PUBLISHED MONTH: **August** YEAR: **2025** 4. FIN OR GRANT NUMBER
5. AUTHOR(S) List names from Cover of NUREG Report of individuals who contributed to this publication	6. TYPE OF REPORT Technical 7. PERIOD COVERED (Inclusive Dates)

8. PERFORMING ORGANIZATION - NAME AND ADDRESS (If NRC, provide Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address; if contractor, provide name and mailing address.)

Performing Organization's Name and Address

9. SPONSORING ORGANIZATION - NAME AND ADDRESS (If NRC, type "Same as above", if contractor, provide NRC Division, Office or Region, U. S. Nuclear Regulatory Commission, and mailing address.)

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Office
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555-0001

10. SUPPLEMENTARY NOTES

First initial. Lastname

11. ABSTRACT (200 words or less)

Copy and paste Abstract here (200 words or less)

12. KEY WORDS/DESCRIPTORS (List words or phrases that will assist researchers in locating the report.)	
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	14. SECURITY CLASSIFICATION *(This Page)* unclassified *(This Report)* unclassified
	15. NUMBER OF PAGES
	16. PRICE

NUREG-XXXX **U.S. NRC Level 3 Probabilistic Risk Assessment Project** **August2025**