ABSTRACT

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| Title of Thesis: | MULTI-UNIT ACCIDENT CONTRIBUTIONS TO U.S. NUCLEAR REGULATORY COMMISSION QUANTITATIVE HEALTH OBJECTIVES: A SAFETY GOAL POLICY ANALYSIS USING MODELS FROM STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES |
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The U.S. Nuclear Regulatory Commission implemented a safety goal policy in response to the 1979 Three Mile Island accident. This policy addresses the question "*How safe is safe enough?*" by specifying quantitative health objectives (QHOs) for comparison with results from nuclear power plant (NPP) probabilistic risk analyses (PRAs) to determine whether proposed regulatory actions are justified based on potential safety benefit. Lessons learned from recent operating experience—including the 2011 Fukushima accident—indicate that accidents involving multiple units at a shared site can occur with non-negligible frequency. Yet risk contributions from such scenarios are excluded by policy from safety goal evaluations—even for the nearly 60% of U.S. NPP sites that include multiple units. This research develops and applies methods for estimating risk metrics for comparison with safety goal QHOs using models from state-of-the-art consequence analyses to evaluate the effect of including multi-unit accident risk contributions in safety goal evaluations.

MULTI-UNIT ACCIDENT CONTRIBUTIONS TO U.S. NUCLEAR REGULATORY
COMMISSION QUANTITATIVE HEALTH OBJECTIVES: A SAFETY GOAL POLICY
ANALYSIS USING MODELS FROM STATE-OF-THE-ART REACTOR
CONSEQUENCE ANALYSES


by


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# List of Abbreviations

AC          alternating current

ACRS        Advisory Committee on Reactor Safeguards

AEA         Atomic Energy Act of 1954, as amended

AEC         U.S. Atomic Energy Commission

ATD         atmospheric transport and dispersion

BDBA        beyond-design-basis accident

BWR         boiling-water reactor

CCDF        complementary cumulative distribution function

CDF         core damage frequency

DBA         design-basis accident

DC          direct current

DOE         U.S. Department of Energy

EDG         emergency diesel generator

EDMG        extensive damage mitigation guideline

EOP         emergency operating procedure

EPZ         emergency planning zone

ERDA        Energy Research and Development Administration

ERP         emergency response plan

FDNPS       Fukushima Daiichi Nuclear Power Station

HEP         human error probability

HRA         human reliability analysis

INES        International Nuclear and Radiological Event Scale

ISLOCA      interfacing systems loss-of-coolant accident

| | |
|---|---|
| LER | licensee event report |
| LERF | large early release frequency |
| LNT | linear no-threshold |
| LOOP | loss of offsite power |
| LTSBO | long-term station blackout |
| LWR | light-water reactor |
| MACCS | MELCOR Accident Consequence Code System |
| NPP | nuclear power plant |
| PCA | probabilistic consequence analysis |
| PRA | probabilistic risk analysis |
| PWR | pressurized-water reactor |
| QHO | quantitative health [effects] objective |
| ORO | offsite response organization |
| RCIC | Reactor Core Isolation Cooling System |
| SAMA | severe accident mitigation alternative |
| SAMDA | severe accident mitigation design alternative |
| SAMG | severe accident mitigation guideline |
| SBO | station blackout |
| SOARCA | State-of-the-Art Reactor Consequence Analyses |
| SSC | structure, system, component |
| STSBO | short-term station blackout |
| TISGTR | thermally-induced steam generator tube rupture |
| USNRC | U.S. Nuclear Regulatory Commission |

# Chapter I: Introduction

## I.A.    Safety Goals for Commercial Nuclear Power Plant Operations

The U.S. Nuclear Regulatory Commission (USNRC) safety goal policy[1] broadly

defines an acceptable level of radiological risk to public health and safety from

potential accidental releases of radiological materials from operating reactor units at

commercial nuclear power plant (NPP) sites. The USNRC developed and evaluated

safety goals in response to growing stakeholder concerns following the March 1979

accident involving Unit 2 of the Three Mile Island Nuclear Station about: (1) the

adequacy of NPP safety; and (2) whether the costs of regulatory actions

implemented in response to the accident to further enhance NPP safety were

justified based on the incremental safety benefit and the level of residual risk to the

public from nuclear accidents. The safety goal policy communicates one aspect of

the USNRC's risk management philosophy; in essence, it addresses the question

"*How safe is safe enough?*" for regulatory decisions regarding NPP safety. In

practice, it guides agency evaluations to determine whether proposed regulatory

actions that would impose additional requirements to enhance NPP safety beyond

those needed to ensure adequate protection are justified based on their potential

safety benefit, relative to the level of residual risk to the public. Using this approach,

proposed regulatory actions that aim to enhance NPP safety can be rejected—even

if potentially cost-beneficial—because the level of residual risk to the public is

acceptably low, and limited resources could thus be better applied to alternative

courses of action.[2,3]

The safety goal policy is based on a hierarchical framework comprised of two high-

level qualitative safety goals supported by two lower-level quantitative health [effects]

objectives (QHOs) that can be used to determine whether and to what extent each qualitative safety goal has been achieved. The first qualitative safety goal addresses risks to individual members of the public and is supported by a QHO for average individual early [prompt] fatality risk (hereafter termed the "*early fatality risk QHO*"). The second qualitative safety goal is intended to address societal risk, but is supported by a QHO for average individual latent cancer fatality risk (hereafter termed the "*latent cancer fatality risk QHO*"). The qualitative safety goals and supporting QHOs are summarized in Table I.

**Table I. USNRC Qualitative Safety Goals and Quantitative Health Objectives[1]**

| Hierarchy Level | Type of Risk Addressed | |
| --- | --- | --- |
| | Individual Risk | Societal Risk |
| High-Level Qualitative Safety Goal | *"Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health."* | *"Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks."* |
| Lower-Level Supporting Quantitative Health Objective | *"The risk to an average individual in the vicinity of a nuclear power plant [a] of prompt fatalities that might result from reactor accidents should not exceed 0.1% of the sum of prompt fatality risks resulting from other accidents to which the members of the U.S. population are generally exposed."* | *"The risk to the population in the area near a nuclear power plant [c] of cancer fatalities that might result from nuclear power plant operation should not exceed 0.1% of the sum of cancer fatality risks resulting from all other causes."* |
| [a] The USNRC safety goal policy statement defines "*in the vicinity of a nuclear power plant*" as the area within one mile of a NPP site boundary. | | |
| [c] The USNRC safety goal policy statement defines "*in the area near a nuclear power plant*" as the area within ten miles of a NPP site boundary. | | |

## I.B. Probabilistic Analysis Techniques

In the nuclear industry, two principal analytic techniques have been used to estimate measures of average individual early fatality risk and latent cancer fatality risk for comparison to the safety goal QHOs: (1) probabilistic risk analysis (PRA); and (2) probabilistic consequence analysis (PCA). PRA is a systematized analytic technique that can be used to characterize the risk of one or more adverse outcomes of interest caused by potential failures involving complex engineered systems—including NPPs. The traditional scenario-based approach to PRA involves systematic application of

methods, models, data, and analytical tools to develop answers to three fundamental

questions that underlie a widely accepted quantitative definition of risk: (1) "*What can

go wrong?*"; (2) "*How likely is it to occur?*"; and (3) "*If it does occur, what are the

consequences?*".[4] In this framework, the risk attributable to accidents caused by

potential failures involving complex engineered systems is characterized by an

ordered set of triplets comprised of scenarios, frequencies, and consequences. In

performing a PRA, the goal is for this set of "*risk triplets*" to capture a reasonably

complete spectrum of possible accident scenarios to provide reasonable assurance

that important risk contributors are not missed.[4]

PCA is used to estimate the third element of the risk triplet by quantifying conditional

measures of the offsite public health, environmental, and economic consequences,

conditioned on the occurrence of a postulated accidental release of radiological

materials from a nuclear facility. PCAs can be performed either as part of Level 3

PRAs for NPPs, or independently for other purposes. In NPP Level 3 PRAs, the

output of PRA logic models that estimate the frequencies of a representative set of

radiological release categories intended to capture a reasonably complete spectrum

of possible accident scenarios is typically combined with the mean[b] conditional PCA

results for each release category.[5] For each outcome of interest, these frequency-

weighted mean consequences are then summed across all radiological release

categories to estimate the mean annual risk of that outcome. Results from NPP

Level 3 PRAs and from PCAs performed for other purposes can include measures

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[b] Common PCA tools for NPP applications can use probabilistic sampling techniques to account for what has been shown in previous studies to be a dominant contributor to uncertainty in the offsite public health consequences for a given accidental release—when the accident will occur and what the prevailing weather conditions will be for the duration of the release(s). Using these probabilistic sampling techniques, PCA tools can generate distributions for selected consequence metrics that reflect variability in consequences arising from statistical variability in weather conditions; the mean conditional PCA results then represent the probability-weighted average for the selected consequence metric over all modeled weather conditions.

of: (1) average individual early fatality risk within one mile of the NPP site boundary; and (2) average latent cancer fatality risk within ten miles of the NPP site boundary. These results have historically been compared to the corresponding safety goal QHOs to: (1) determine whether and to what extent the qualitative safety goals have been achieved; or (2) provide an additional perspective or reference point for interpreting results within a safety goal QHO context.

## I.C.    Research Motivation

In the U.S., a majority of NPP sites include multiple operating reactor units co-located at a shared site. As shown in Figure 1, 52% (32 out of 61) of U.S. NPP sites include two operating reactor units and 5% (3 out of 61) include three operating reactor units;[6] nearly 60% of all U.S. NPP sites are thus multi-unit sites. However, during development and evaluation of the safety goal policy, the USNRC decided that the safety goals and QHOs would be applied strictly on a per-reactor-unit basis, even for this majority of multi-unit sites; the rationale for this decision was to avoid imposing a regulatory bias against multi-unit sites that may be subject to stricter requirements if the safety goals and QHOs were to be applied on a per-site basis.[7] As a result, accident scenarios involving concurrent radiological releases from multiple co-located operating reactor units have—with few exceptions—traditionally been excluded from NPP PRAs and safety goal evaluations in support of analyses for proposed regulatory actions.

Yet there are at least three compelling reasons for expanding the scope of NPP PRAs and safety goal evaluations to include consideration of multi-unit accident scenarios:

1. **Previous NPP PRA insights:** Findings from previous NPP PRAs that included a limited treatment of multi-unit accident scenarios (e.g., the Seabrook Station Probabilistic Safety Assessment)[8] suggest that the contribution to reactor accident risk from such scenarios is not negligible and could be significant, depending on site-specific factors that influence the potential for dependent failure events or adverse interactions across multiple units.[c]

2. **Operating experience:** Lessons learned from the March 2011 accident at the Fukushima Daiichi Nuclear Power Station (FDNPS) in Japan[9-11]—together with findings from a recent review of licensee event report (LER) system data for U.S. NPPs[12,13]—demonstrate that adverse events (e.g., abnormal occurrences, incidents, or accidents) involving multiple operating reactor units co-located at a shared site can and do occur at a non-negligible frequency.[d]

3. **Logical reasoning:** Table I shows that people living in the vicinity of or area near NPP sites are the defined target population for the safety goals and QHOs.[1] For multi-unit NPP sites, this population is not only exposed to the health and safety

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[c] In probability theory, two events A and B are considered to be dependent when the probability of their joint occurrence (termed "*the intersection of A and B,*" which is mathematically represented by the term "$A \cap B$") is not equal to the product of the probabilities for each event. The equivalent mathematical statement if A and B are dependent events is: $P(A \cap B) = P(A) \cdot P(B|A) = P(B) \cdot P(A|B) \neq P(A) \cdot P(B)$. For dependent events, this inequality arises from dependencies that cause the conditional probability of an event, given the occurrence of another event, to be different from the event's unconditional probability. The equivalent mathematical statements for the given example are: $P(A|B) \neq P(A)$ and $P(B|A) \neq P(B)$. Where dependencies exist, these can reduce or eliminate redundancy in design. As a result, dependencies usually cause the probability of the joint occurrence of failure events to be greater than the product of the probabilities of independent failure events. For this reason, dependencies between multiple reactor units at a shared NPP site are an important consideration in evaluating the contribution to risk from multi-unit accident scenarios.

[d] Based on the recent review of LER system data, Schroer developed a dependent failure event classification scheme to characterize potential dependencies across multiple units so that single-unit PRA models could to be integrated into a multi-unit PRA model for a shared NPP site. This classification scheme is comprised of six different categories of dependencies that could influence multi-unit risk: (1) initiating events, (2) shared connections, (3) identical components, (4) proximity dependencies, (5) human dependencies, and (6) organizational dependencies.[12,13]

risks posed by single-unit accident scenarios in which isolated accidents involving single operating reactor units occur; they are also exposed to the potentially significant health and safety risks from possible multi-unit accident scenarios that operating experience has shown can and do occur. It therefore stands to reason that the contributions to risk from such multi-unit accident scenarios should be considered in safety goal policy applications.
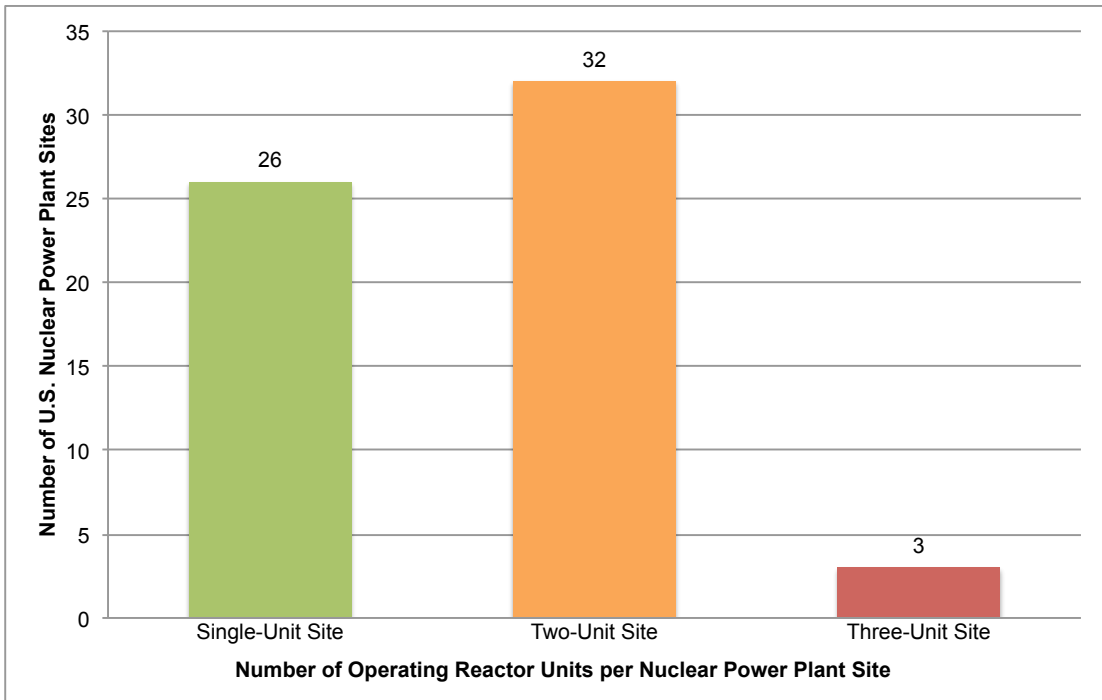


**Figure 1. Discrete Frequency Distribution of U.S. Nuclear Power Plant Sites by Number of Operating Reactor Units per Site.** Nearly 60% of U.S. NPP sites are multi-unit sites comprised of two or three operating reactor units.

This existing gap in the scope and application of the safety goal policy can have important implications. In particular, since multi-unit accident scenarios are excluded from the scope of NPP PRAs and supporting PCAs, the true total accident risk for reactor units that are co-located with other units at multi-unit NPP sites may be underestimated. As a result, safety goal evaluations of proposed regulatory actions that aim to further enhance NPP safety beyond the level provided by adequate

protection—and thereby reduce the overall residual risk to public health and safety from NPP accidents—may conclude that these proposed actions would not be justified based on a low level of residual risk to the public. These proposed actions would therefore be rejected before detailed cost-benefit analyses would be performed to determine whether this subset of decision analysis techniques indicate the actions could result in a net benefit to society and thus improve societal welfare.

## I.D.    Research Aims

The overall aim of this research is to evaluate the potential policy implications of expanding the scope and application of the safety goal policy to include consideration of the risk contribution from multi-unit accident scenarios for multi-unit NPP sites. From this overall research aim, three specific aims were developed to guide this investigation:

1. **Specific Aim 1:** To evaluate the effect of including the contribution from multi-unit accident scenarios to safety goal QHO risk metrics in safety goal policy applications. For this aim, base case analyses were performed that relied on two assumptions that each influence one element of the risk triplet for multi-unit accident scenarios:

    a. *Frequency:* The conditional probability of an accident occurring in a co-located unit, given that an accident involving at least one unit at the NPP site occurs at a specified frequency, was assumed to be 0.1. This implies a 10% chance of a co-located unit experiencing an accident scenario, given that single-unit accident scenarios occur at the site with a specified frequency. Results and insights from previous multi-unit NPP PRA studies and operating experience data suggest this is a reasonable assumption. In addition, a

global average conditional probability of 0.1 is assumed to apply across all multi-unit accident scenarios. In reality, each multi-unit accident scenario can have a unique conditional probability given the occurrence of a specified single-unit accident scenario.

b. *Consequences:* Multi-unit accident scenarios were assumed to occur simultaneously, with no timing offset between concurrent accidents involving multiple units. The hypothesis underlying this assumption is that simultaneous accidents will result in the most severe offsite consequences, though this may not be the case for all consequence metrics of interest.

2. **Specific Aim 2:** To evaluate the effect on findings from Specific Aim 1 of using plausible alternative assumptions about the level of dependence between co-located reactor units at a shared NPP site. For this aim, one-way sensitivity analyses were performed to evaluate the effect of varying the conditional probability of an accident in a co-located unit given a single-unit accident frequency over a range of plausible values.

3. **Specific Aim 3:** To evaluate the effect on findings from Specific Aim 1 of using plausible alternative assumptions about the timing of multi-unit accident scenarios and the constituent releases that comprise each multi-unit release. For this aim, one-way sensitivity analyses were performed to evaluate the effect of varying a timing offset parameter over a range of plausible values. This parameter is used to represent potential differences in the timing of accident initiation, progression, and radiological releases between concurrent accident scenarios involving multiple reactor units co-located at a shared NPP site.

## I.E. Research Study Design

### I.E.1. Safety Goal Policy Alternatives

Two safety goal policy alternatives were selected for this evaluation:

1. **Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included.** This option represents the status quo with respect to how the safety goal policy is applied on a per-reactor unit basis. Only the contribution from single-unit accident scenarios is included in estimating the safety goal QHO risk metrics for comparison to the QHOs.

2. **Option 2: Single-Unit and Multi-Unit Accident Scenarios Included.** This option represents a hypothetical expansion in scope and application of the safety goal policy in applying the policy on a per-reactor unit basis. The contribution from both single-unit accident scenarios and multi-unit accident scenarios is included in estimating the safety goal QHO risk metrics for comparison to the QHOs.

### I.E.2. Figures of Merit

Two figures of merit based on the safety goal QHO risk metrics for average individual early fatality risk and average individual latent cancer fatality risk were selected to evaluate the effects of this expansion in scope:

1. **Figure of Merit 1: Percent change in mean[e] safety goal QHO risk metric results.** The first figure of merit is the percent change in the mean value for each

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[e] As stated earlier, PCA codes—including the one used in this research—are capable of estimating the probability distribution for each consequence metric it calculates. These distributions reflect the uncertainty in consequence results arising from statistical variability in the offsite weather conditions and when in a year an accident scenario will occur. Consistent with guidance provided in the USNRC Safety Goal Policy Statement,[1] only estimates of the mean values for these distributions will be used for this evaluation

safety goal QHO risk metric when comparing Option 2 results to Option 1 results. This provides a measure of the impact of including the risk contribution from multi-unit accident scenarios on each safety goal QHO risk metric.

2. **Figure of Merit 2: Percent change in mean QHO margin.** The second figure of merit is the percent change in the mean value for the margin to each QHO when comparing Option 2 results to Option 1 results. This provides a measure of the impact of including the risk contribution from multi-unit accident scenarios on the margin to each QHO.[f]

### I.E.3.  *Study Population and Accident Scenarios*

This research builds upon a recent USNRC-sponsored PCA study that is commonly referred to as the "*State-of-the-Art Reactor Consequence Analyses (SOARCA) Project*."[14-16] The SOARCA pilot study developed state-of-the-art accident progression and offsite radiological consequence models to characterize realistic outcomes of a select set of single-unit accident scenarios that were judged to be more important. For the pilot study, detailed models and integrated analyses were performed for single operating reactor units at two representative U.S. NPP sites that are each comprised of two operating reactor units: (1) Peach Bottom Atomic Power Station, Unit 2 (hereafter termed "*Peach Bottom*"); and (2) Surry Power Station, Unit 1 (hereafter termed "*Surry*").[g] The single-unit accident scenarios were selected based on expert judgments about their relative importance with respect to: (1) the likelihood of causing damage to the nuclear fuel in the reactor core (termed "*core*

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[f] The margin to each QHO can be viewed as the relative distance between the QHO and the value for the corresponding safety goal QHO risk metric. This type of measure can provide information about the factor by which the value of a safety goal QHO risk metric would need to increase to reach the corresponding QHO.

[g] Multi-unit accident scenarios involving co-located reactor units at each NPP site (Unit 3 at Peach Bottom and Unit 2 at Surry) were excluded from the scope of the SOARCA pilot study.

*damage*"); or (2) the potential to cause significant offsite radiological health consequences, assuming each accident scenario were to occur. A summary of the single-unit accident scenarios evaluated for each NPP site in the SOARCA pilot study is provided in Table II and Table III.

To address the aims for this research, state-of-the-art consequence models were constructed for all possible combinations of two-unit accident scenarios that could be created by combining the single-unit accident scenarios that were modeled and evaluated for each NPP site in the SOARCA pilot study. These two-unit accident scenario models were developed and evaluated using the MELCOR Accident Consequence Analysis Code System (MACCS) suite of analytical tools. MACCS is a USNRC-sponsored PCA code that integrates probabilistic and phenomenological models to account for multiple factors that influence the offsite consequences of accidental releases, including: (1) statistical variability in weather conditions over time; (2) atmospheric transport and dispersion of released radiological materials; (3) offsite population characteristics; (4) protective actions taken to reduce radiological dose; and (5) dose-response models used to estimate radiological health effects.[17,18]

MACCS was recently enhanced to include a multi-source model that allows users to model and analyze concurrent accidental releases from multiple co-located units at a shared nuclear facility that can have unique accident progression timelines and radionuclide inventories. Event trees and decision trees are useful tools for illustrating the possible combinations that can arise from combining sequential events and/or decisions. Figures 2 and 3 respectively show that: (1) nine two-unit accident scenario models for Peach Bottom Unit 2 and Unit 3[h] can be created from all possible combinations of the three single-unit accident scenarios evaluated for

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[h] Peach Bottom Unit 1 completed decommissioning in 1978 and is no longer operational.

Peach Bottom as part of the SOARCA pilot study; and (2) 16 two-unit accident

scenario models for Surry Unit 1 and Unit 2 can be created from all possible

combinations of the four single-unit accident scenarios evaluated for Surry. Together,

this results in a total of 25 two-unit accident scenario models across both

representative NPP sites.

For each of these 25 two-unit accident scenario models, MACCS was used to

perform eight discrete probabilistic accident simulations to calculate the conditional

consequence contribution to safety goal QHO risk metrics. One simulation

represented the base case analysis that assumed the constituent accident scenarios

that comprise each two-unit accident scenario occur simultaneously. The remaining

seven simulations represented sensitivity cases for two-unit accident scenarios in

which the timing offset between concurrent accidents involving both units was varied

from 1 to 7 days, in one-day increments.[i] The purpose of these one-way sensitivity

analyses was to evaluate the effect on results of using plausible alternative

assumptions about the potential difference in timing between accident initiation,

progression, and radiological releases across co-located units. Collectively, this

resulted in a total of 200 two-unit accident simulations across both NPP sites

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[i] A limited set of sensitivity analyses were also performed to evaluate the effect of varying the timing offset parameter from 1 to 24 hours in one-hour increments. Only results for one-way sensitivity analyses using the 1 to 7 day range in one-day increments are presented for two reasons: (1) results did not converge within 24 hours for some two-unit accident scenarios, indicating the need to perform sensitivity analyses using timing offsets greater than 24 hours; and (2) one-day increments appeared to provide sufficient resolution to characterize patterns or trends.

**Table II. Single-Unit Accident Scenarios Evaluated in the SOARCA Pilot Study[14-16]**

| NPP Site | Accident Scenario 1 | Accident Scenario 2 | Accident Scenario 3 | Accident Scenario 4 |
|---|---|---|---|---|
| Peach Bottom | LTSBO | STSBO–Base | STSBO–RCIC | N/A |
| Surry | LTSBO | STSBO–Base | STSBO-TISGTR | ISLOCA |

**Table III. Single-Unit Accident Scenario Descriptions for the SOARCA Pilot Study[14-16]**

| Acronym | Title | Description |
|---|---|---|
| SBO | Station Blackout | NPP safety systems are powered by alternating current (AC) power. This ac power is normally supplied by offsite power sources via the electrical grid, but can be supplied by onsite backup power sources such as emergency diesel generators, if needed. An SBO involves the total loss of AC power that results when both offsite and onsite AC power sources fail. During an SBO, reactor cooling is temporarily provided by systems that do not rely on AC power, such as pumps driven by steam turbines. Onsite batteries can temporarily supply direct current (DC) power to control these turbine-driven pumps and to power instrumentation until battery depletion. |
| LTSBO | Long-Term Station Blackout | An earthquake causes a loss of all AC power sources, but onsite batteries are able to supply DC power to safety systems for about 4-8 hours until battery depletion. |
| STSBO | Short-Term Station Blackout | An earthquake more extreme than the LTSBO scenario earthquake causes a total loss of all AC and DC power sources, immediately rendering safety systems inoperable. As a result, onset of damage to nuclear fuel in the reactor core occurs in the "short-term." This base case is identified as "*STSBO-Base*" in Table II. |
| RCIC | Reactor Core Isolation Cooling System | This scenario-identified as "*STSBO-RCIC*" in Table II-is a variation of the STSBO that applies only to BWR NPPs, which include the RCIC system. This scenario was selected for evaluation because the modeled NPP site (Peach Bottom) had explicit procedures for operating the RCIC system using portable electric generators in SBO conditions to provide reactor cooling. |
| TISGTR | Thermally-Induced Steam Generator Tube Rupture | This scenario-identified as "*STSBO-TISGTR*" in Table II-is a lower probability variation of the STSBO that applies only to PWR NPPs, which include steam generators for steam production. While the reactor core is overheating and water available for heat transfer in the steam generators is boiling off, extremely hot steam and hydrogen circulating through the steam generator cause a tube to rupture; this creates a pathway for radiological materials to escape from the reactor coolant system to the NPP's non-radiological systems, and potentially to the environment. |
| ISLOCA | Interfacing Systems Loss-Of-Coolant Accident | A random failure of valves ruptures low-pressure system piping outside the containment building that connects with the high-pressure reactor coolant system piping that is inside the containment building. This failure bypasses the defense-in-depth layer of protection provided by the containment building, thereby resulting in a more rapid radiological release to the environment, with greater potential for causing fatalities among the offsite population. |

| ID No. | Two-Unit Accident Scenario |
|---|---|
| **BWR1** | Unit 2: LTSBO, Unit 3: LTSBO |
| | |
| **BWR2** | Unit 2: LTSBO, Unit 3: STSBO-Base |
| **BWR3** | Unit 2: LTSBO, Unit 3: STSBO-RCIC |
| **BWR4** | Unit 2: STSBO-Base, Unit 3: LTSBO |
| | |
| **BWR5** | Unit 2: STSBO-Base, Unit 3: STSBO-Base |
| **BWR6** | Unit 2: STSBO-Base, Unit 3: STSBO-RCIC |
| **BWR7** | Unit 2: STSBO-RCIC, Unit 3: LTSBO |
| | |
| **BWR8** | Unit 2: STSBO-RCIC, Unit 3: STSBO-Base |
| **BWR9** | Unit 2: STSBO-RCIC, Unit 3: STSBO-RCIC |
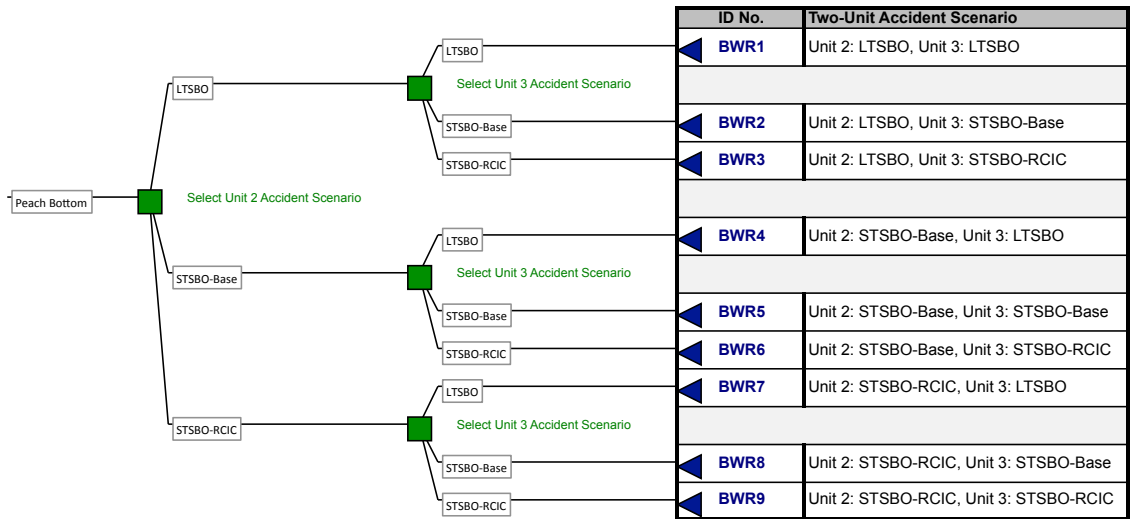
**Figure 2. Two-Unit Accident Scenario Models for Peach Bottom Unit 2 and Unit 3.** Nine two-unit accident scenario models were constructed from all possible combinations of the three single-unit accident scenario models for Peach Bottom Unit 2 that were evaluated in the SOARCA pilot study.



| ID No. | Two-Unit Accident Scenario |
|---|---|
| **PWR1** | Unit 1: LTSBO, Unit 2: LTSBO |
| **PWR2** | Unit 1: LTSBO, Unit 2: STSBO-Base |
| | |
| **PWR3** | Unit 1: LTSBO, Unit 2: STSBO-TISGTR |
| **PWR4** | Unit 1: LTSBO, Unit 2: ISLOCA |
| **PWR5** | Unit 1: STSBO-Base, Unit 2: LTSBO |
| **PWR6** | Unit 1: STSBO-Base, Unit 2: STSBO-Base |
| | |
| **PWR7** | Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR |
| **PWR8** | Unit 1: STSBO-Base, Unit 2: ISLOCA |
| **PWR9** | Unit 1: STSBO-TISGTR, Unit 2: LTSBO |
| **PWR10** | Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base |
| | |
| **PWR11** | Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR |
| **PWR12** | Unit 1: STSBO-TISGTR, Unit 2: ISLOCA |
| **PWR13** | Unit 1: ISLOCA, Unit 2: LTSBO |
| **PWR14** | Unit 1: ISLOCA, Unit 2: STSBO-Base |
| | |
| **PWR15** | Unit 1: ISLOCA, Unit 2: STSBO-TISGTR |
| **PWR16** | Unit 1: ISLOCA, Unit 2: ISLOCA |

**Figure 3. Two-Unit Accident Scenario Models for Surry Unit 1 and Unit 2.** Sixteen two-unit accident scenario models were constructed from all possible combinations of the four single-unit accident scenario models for Surry Unit 1 that were evaluated in the SOARCA pilot study.

## I.F.    Key Assumptions

This study design relies on a number of key assumptions:

1. **The modeled NPP sites are representative of the population of multi-unit NPP sites.** The two NPP sites selected for modeling and evaluation as part of this research constitute only 6% (2 out of 35) of the population of multi-unit NPP sites in the U.S. However, these NPP sites utilize reactor and containment designs similar to those used at sites that collectively represent 74% (26 out of 35) of U.S. multi-unit NPP sites. This analysis therefore assumes that the two modeled NPP sites are representative of the population of U.S. multi-unit NPP sites in any attempts to generalize its findings beyond the study population.

2. **The two operating reactor units co-located at each NPP site are identical.** The consequence model for each two-unit accident scenario is constructed by combining the radiological release inputs to the consequence models from two single-unit accident scenarios that were modeled in the SOARCA pilot study. In this approach, the first single-unit accident scenario represents the accident scenario that occurs in the reference unit[j], while the second represents the concurrent accident scenario that occurs in the co-located unit. Since all of the models for the SOARCA pilot study were based on one unit at each NPP site (Unit 2 at Peach Bottom and Unit 1 at Surry), use of these models to represent concurrent accident scenarios at the co-located units at each NPP site (Unit 3 at Peach Bottom and Unit 2 at Surry) implicitly assumes the co-located units are identical to the units modeled in the SOARCA pilot study. While safety analysis reports for each site indicate this assumption of symmetry is reasonable, there

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[j] The need for identifying which unit serves as the reference unit for the two-unit accident scenarios is explained next with Key Assumption #3.

can be subtle differences between the co-located units that can lead to biased estimates of their risk contributions. In fact, this issue is not unique to the Peach Bottom and Surry NPP sites; while co-located units at shared NPP sites in the U.S. are typically similar in many respects, it is widely recognized that nearly all co-located units have some differences in design and operation.

3. **One unit always serves as the reference unit for two-unit accident scenarios.** For each two-unit accident scenario, one unit at each NPP site is assumed to be the reference unit. For Peach Bottom, the reference unit is Unit 2; for Surry, the reference unit is Unit 1. This assumption simplifies the analysis by requiring the analyst to specify a value for only one additional parameter in implementing the new multi-source model in MACCS; this parameter specifies the timing offset between the releases from the co-located unit, relative to those from the reference unit. This assumption therefore does not impact the base case analyses in which two-unit accident scenarios are assumed to occur simultaneously. However, for one-way sensitivity analyses in which the timing offset between concurrent accidents involving both units is varied, this assumption means the reference unit's accident scenario will always progress ahead of the co-located unit's accident scenario. If the assumption of identical units holds, this assumption will have no effect on the results and findings. However, this assumption could introduce bias if there are factors that result in differences in risk contributions based on which unit's accident scenario is initiated and progresses first.

4. **The consequence models from the SOARCA pilot study are valid.** The state-of-the-art consequence models developed for each of the single-unit accident scenarios evaluated as part of the SOARCA pilot study serve as the foundation

for the consequence models developed for each two-unit accident scenario evaluated as part of this research. The technical bases for modeling assumptions and parameter values used in the SOARCA pilot study are well-documented and have undergone extensive peer review.[14-16] However, any bias introduced by modeling choices made in the SOARCA pilot study may bias the results of this research.

5. **The accident scenarios modeled and evaluated are representative of the full spectrum of potential accident scenarios for each NPP site.** The safety goal QHOs were developed for comparison with safety goal QHO risk metric results from full-scope NPP Level 3 PRAs that model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential accident scenarios. To evaluate the effect of expanding the scope of the safety goal policy to include consideration of the risk contribution from multi-unit accident scenarios, this analysis assumes that the limited set of single-unit and two-unit accident scenarios modeled for each NPP site is representative of the full spectrum of potential accident scenarios that could occur at each site with respect to their conditional consequence contribution to the safety goal QHO risk metrics. This assumption allows for adjusting the results of each accident scenario using a frequency adjustment factor to account for the frequency contribution to risk from excluded scenarios that belong to representative classes to develop an estimate of total accident risk. However, this assumption can lead to biased results if there are accident scenarios that are not adequately represented by the modeled set—especially if they result in significantly different conditional consequence contributions to the safety goal QHO risk metrics.

## I.G. Analysis Scope

As stated in Section I.F., the safety goal QHOs were developed for comparison with safety goal QHO risk metric results from full-scope NPP Level 3 PRAs that model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential accident scenarios. A full-scope NPP Level 3 PRA can include accident scenarios that: (1) are initiated by internal and external initiating event hazards; (2) can occur while the NPP is in different plant operating states (e.g., at-power, low-power, or shutdown); and (3) involve other major sources of radiological materials on the NPP site (e.g., spent fuel pool units or dry cask storage facilities).[5]

This research is limited to the set of seven single-unit accident scenarios that were modeled and evaluated as part of the SOARCA pilot study and the set of 25 two-unit accident scenarios that could be constructed from all possible combinations of these single-unit accident scenarios. As a result, the following groups of accident scenarios that could potentially be included in a full-scope NPP Level 3 PRA are explicitly excluded from the scope of this analysis:

1. **Reactor accident scenarios initiated by deliberate malevolent acts.** Deliberate malevolent acts include acts of sabotage and terrorist attacks. Reactor accident scenarios initiated by these types of acts are typically excluded from full-scope NPP Level 3 PRAs and are also considered to be beyond the scope of the safety goal policy.

2. **Reactor accident scenarios that occur while the reactor is in plant operating states other than at-power.** These include reactor accident scenarios that occur while the reactor is in low-power or shutdown plant operating states.

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3. **Accident scenarios involving other major sources of radiological materials at the modeled NPP sites.** These include accident scenarios involving the spent fuel pool units or dry cask storage facilities at each of the modeled NPP sites.

## I.H.    Significance of Research and Potential Policy Implications

This research makes three novel and significant contributions:

1. It develops efficient methods for estimating the contributions to safety goal QHO risk metrics from classes of single-unit and multi-unit accident scenarios using realistic consequence models from contemporary state-of-the-art reactor consequence analyses that leveraged decades of severe accident research and advanced analytical tools. These methods were demonstrated using a two-unit case study involving two NPP sites that are considered to be representative of a broad class of multi-unit U.S. NPP sites that utilize similar reactor and containment designs.

2. It develops and evaluates state-of-the-art consequence models for concurrent multi-unit accident scenarios involving both: (1) simultaneous accidents in multiple units; and (2) staggered accidents in which the timing offset between concurrent accidents involving multiple co-located units at a shared NPP site is varied over a range of plausible alternative values.

3. It generates new insights about: (1) the relative contributions of single-unit and multi-unit accident scenarios to safety goal QHO risk metrics; and (2) the effect on safety goal QHO risk metrics and margin to the QHOs if the risk contribution from multi-unit accident scenarios were to be included in safety goal policy evaluations.

Insights derived from this research could be used to inform current USNRC and nuclear industry stakeholder deliberations about whether and to what extent the existing safety goal policy should be expanded to include consideration of the risk contribution from multi-unit accident scenarios. Such an expansion in the scope of the safety goal policy could potentially yield different decisions regarding the justification of future proposed regulatory actions that aim to further enhance NPP safety beyond the level provided by adequate protection and thereby reduce the overall residual risk to public health and safety. In particular, this could have significant implications for regulatory requirements, policies, or guidance pertaining to defense-in-depth elements designed to limit the public health and safety risks due to potential accidents involving operating reactor units at multi-unit NPP sites; notable examples include:

1. **Design of NPP Structures, Systems, Components (SSCs) and Severe Accident Management Guidelines (SAMGs) for Multi-Unit NPP Sites.** NPPs are equipped with SSCs designed to prevent, delay, or limit the amount of radiological material released to the surrounding environment—and thus limit the public health and safety consequences—if a severe accident involving core damage were to occur. SAMGs provide flexible guidance regarding the use of a set of potential accident management strategies designed to stop the progression of core damage and to limit the radiological release to the environment.

2. **Multi-Unit NPP Siting Requirements.** USNRC regulations specify criteria for siting nuclear reactors. For multi-unit NPP sites with multiple interconnected reactors, these regulations specify that siting requirements shall be based upon the assumption that all interconnected reactor units release radiological materials

to the environment simultaneously;[19] this implicitly assumes that the simultaneous release is a bounding or worst-case scenario, which may not be the case for all consequence metrics of interest in NPP siting applications.

3. **Multi-Unit NPP Site Emergency Planning Requirements.** USNRC regulations require NPP licensees to develop detailed emergency response plans (ERPs) for specified emergency planning zones (EPZs) around the NPP site to ensure that preplanned protective actions can be taken to adequately protect the public in the event of a severe accident.[20] In establishing these regulatory requirements, the USNRC relied on results from NPP Level 3 PRA and PCA studies that did not consider multi-unit accident scenarios for NPP sites with multiple co-located reactor units.

## Chapter II: Background

### II.A.    Probabilistic Analysis Techniques for Nuclear Power Plants

Effectively managing the risk of adverse human health, environmental, and economic consequences posed by possible nuclear accident scenarios first requires accurate identification and assessment of these risks. PRA is a subset of risk analysis techniques commonly used to support risk management or safety-related decisions involving complex engineered systems. The traditional scenario-based approach to PRA involves systematic application of methods, models, data, and analytic tools to develop answers to three fundamental questions that underlie a widely accepted quantitative definition of risk:[4]

1. **"*What can go wrong?*"** To answer this question, analysts identify a set of hazards that pose a threat to the modeled system or its environment and develop a set of possible accident scenarios. Each accident scenario begins with an initiating event that causes the system to deviate from its intended operating state and challenges mitigating systems, and ends with an adverse outcome or end state of interest. Within each scenario, there can be multiple intermediate events—such as successes or failures of engineered safety features or human actions—that determine whether and how an initiating event leads to a particular end state.

2. **"*How likely is it to occur?*"** To answer this question, analysts estimate the likelihood of each accident scenario using its initiating event frequency and the conditional probabilities of different combinations of intermediate events that can lead from the initiating event to a particular end state, assuming the initiating event occurs.

3. **"*If it does occur, what are the consequences?*"** To answer this question, analysts estimate the conditional consequences resulting from each accident scenario, assuming it occurs. This requires specification and estimation of consequence measures that represent the level of damage or loss that can occur in terms of the adverse outcomes of interest. In general, there are four major categories of consequence measures, organized by the type of adverse outcome they are designed to measure: (1) dose levels resulting from population exposure to harmful agents—including ionizing radiation; (2) adverse human health effects resulting from exposure to harmful agents; (3) adverse environmental impacts; and (4) economic damages or financial loss.

The risk triplet comprised of an accident scenario, its frequency, and its conditional consequences represents the risk attributed to a particular class of scenarios. In this traditional scenario-based PRA framework, the ordered set of risk triplets that encompasses a reasonably complete spectrum of possible accident scenarios that can occur is assumed to represent the total risk attributed to postulated accidents caused by failures within the modeled system.

PRA has been used primarily as a decision support tool to inform risk management or safety-related decisions involving complex engineered systems. In particular, PRA has been successfully applied to a vast array of technological systems to: (1) identify vulnerabilities and interdependencies in system design and performance that increase the risk of system failures; (2) characterize the risk of adverse human health, environmental, and economic consequences attributable to possible system failures and accidents; (3) identify and characterize significant contributors to risk; (4) characterize the relative effectiveness of alternative actions or system configurations for reducing risk; and (5) prioritize the allocation of scarce resources to selected risk

reduction measures or safety enhancements.[21,22] In this way, PRA has provided valuable insights that—together with other analytic techniques—result in better-informed risk management or safety-related decisions involving complex engineered systems.

Within this scenario-based approach to PRA, risk can be characterized in many ways, depending on the end states of interest for a particular decision or application. To provide some overall logic and structure and to facilitate evaluation of intermediate results, PRAs for NPPs have traditionally been organized into three analysis levels, with the scope and level of complexity of the PRA model increasing with each level. These levels are defined by three sequential adverse end states that can occur in the progression of postulated NPP accident scenarios: (1) onset of core damage; (2) release of radioactive materials from the NPP containment structure to the surrounding environment (termed "*radiological release*"); and (3) adverse human health, environmental, and economic consequences that occur beyond the boundary of the NPP site (commonly grouped into the broad term "*offsite radiological consequences*").[5]

Figure 4 illustrates the overall logic and structure of traditional NPP PRA models, including the types of results that are produced at each level. As shown, the end state of interest for a Level 1 PRA is core damage. A Level 1 PRA model therefore estimates core damage frequency (CDF) using linked event tree and fault tree logic models that represent initiating events and response of mitigating systems. The end state of interest for a Level 2 PRA is radiological release. A Level 2 PRA model therefore expands upon a Level 1 PRA model by adding severe accident phenomenological models and logic models that represent containment systems response to estimate radiological release category frequencies and various

characteristics of the released radioactive material (commonly referred to as the "*source term*"). Finally, the end states of interest for a Level 3 PRA are various offsite radiological consequences. A Level 3 PRA model therefore expands upon a Level 2 PRA model by adding a PCA element to quantify conditional measures of the offsite radiological health, environmental, and economic consequences, conditioned on the occurrence of each postulated radiological release category and its representative source term that provides input to the offsite radiological consequence model.[5]

Increasing PRA model scope and complexity →

| LEVEL 1 PRA MODEL | LEVEL 2 PRA MODEL | LEVEL 3 PRA MODEL |
|---|---|---|
| Initiating Events & Mitigating Systems Response Logic Models (Linked Event Trees & Fault Trees) | Severe Accident Phenomenological Models & Containment Systems Response Logic Models | Conditional Probabilistic Consequence Analysis Models |

| Core Damage | Radiological Release | Offsite Radiological Consequences |
|---|---|---|
| • Total core damage frequency<br>• Core damage sequence information<br>• Importance measures | • Radiological release category frequencies<br>• Representative source term information | • Population dose<br>• Adverse human health effects<br>• Contaminated areas<br>• Economic costs |

**Figure 4. Overall Logic and Structure of Traditional NPP PRA Models.** NPP PRA models have traditionally been organized into three analysis levels, with the scope and level of complexity of the PRA model increasing with each level. These levels are defined by three sequential adverse end states that can occur in the progression of postulated NPP accident scenarios: (1) core damage, (2) radiological release, and (3) offsite radiological consequences.

In NPP Level 3 PRAs, the output of PRA logic models that estimate the frequencies of a representative set of radiological release categories intended to capture a reasonably complete spectrum of possible accident scenarios is combined with the conditional PCA results for each release category. For each outcome of interest, the

frequency-weighted mean consequences are then summed across all radiological

release categories to estimate the mean annual risk of that outcome. In addition to

the mean risk of each consequence metric, other quantitative and graphical methods

are commonly used to characterize the public risk attributable to nuclear accidents.

Notable examples include: (1) probabilistic summary measures for consequence

metric distributions (e.g., 95th and 5th percentiles); and (2) complementary

cumulative distribution function (CCDF) curves (also termed "*exceedance frequency*

*curves*," "*risk curves*," or "*risk profiles*") that illustrate the probability or frequency of

exceeding specified consequence levels.[5]

Whether performed as part of a NPP Level 3 PRA, or independently for another

purpose, PCAs are typically used to assess the offsite radiological consequences of

severe or beyond-design-basis accidents (BDBAs).[k] Applications of PCA at the

USNRC include: (1) regulatory analyses[2,3] and backfit[l] analyses[24] to support

decisions regarding proposed regulatory actions; (2) environmental assessment

reviews with respect to severe accidents and Severe Accident Mitigation Alternatives

(SAMA) analyses for operating power reactor license renewal[25] or Severe Accident

Mitigation Design Alternatives (SAMDA) analyses for new power reactor design

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[k] The USNRC defines a design-basis accident (DBA) as: "*a postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety*" (http://www.nrc.gov/reading-rm/basic-ref/glossary/design-basis-accident.html). Beyond-design-basis accidents (BDBAs) are defined as: "*accident sequences that are possible but were not fully considered in the design process because they were judged to be too unlikely.*" BDBAs are thus considered to be beyond the scope of DBAs that a nuclear facility must be designed and built to withstand (http://www.nrc.gov/reading-rm/basic-ref/glossary/beyond-design-basis-accidents.html).

[l] A backfit is a generic or plant-specific modification that becomes effective after specified dates. Examples of backfits include modification of or addition to: (1) facility SSCs or design; (2) the design approval or manufacturing license for a facility; or (3) the procedures or organization required to design, construct or operate a facility. Any of these modifications or additions may result from a new or amended provision in USNRC regulations or the imposition of a regulatory staff position interpreting USNRC regulations that is either new or different from a previously applicable staff position.[23]

stage applications;[26] and (3) supporting applied research studies—including the

SOARCA project,[14-16] which developed much of the technical basis for the state-of-

the-art consequence models used in this research.

## II.B.  USNRC Safety Goal Policy Statement

A natural question that emerged in the late 1970s and early 1980s as some of the

earliest NPP Level 3 PRAs were being completed—and as the USNRC and nuclear

industry were responding to the accident at the Three Mile Island Nuclear

Station—was how to interpret the results from these PRAs and determine whether

they are acceptable. The USNRC's approach to addressing this question was to

develop a safety goal policy. There is a rich history associated with the genesis,

development, evaluation, and implementation of the USNRC policy statement on

safety goals for NPP operations. Since this has been well documented in many

accessible sources,[1,7,27,28] the background discussion provided here is relatively brief

and focuses on the essential issues that are germane to this research.

The USNRC[m] derives its statutory responsibilities and authorities from the Atomic

Energy Act of 1954, as amended (AEA).[30] AEA provisions contain an "*adequate*

*protection standard*" that represents a minimum safety standard the USNRC is

required to satisfy. Under this standard, the USNRC must—at a minimum—ensure

the health and safety of the public are adequately protected in executing its statutory

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[m] The U.S. Atomic Energy Commission (AEC) was an independent agency in the federal executive branch that preceded the USNRC. Under the original Atomic Energy Act of 1954, the AEC was responsible for two basic functions: (1) the promotion and development of commercial uses of nuclear materials; and (2) the licensing and regulation of commercial uses of nuclear materials to ensure adequate protection of public health and safety. Amidst growing public concerns about an inherent conflict of interest between these dual functions of the AEC, Congress enacted legislation to officially separate them. The Energy Reorganization Act of 1974[29] abolished the AEC and created two new agencies: (1) the Energy Research and Development Administration (ERDA), now the U.S. Department of Energy (DOE), which was assigned the AEC's promotion and development functions; and (2) the USNRC, which was assigned the AEC's licensing and regulatory functions.

functions. With respect to risk, achieving adequate protection means NPPs must pose "*no undue risk*" (not zero risk) to public health and safety; this means adequate protection can be achieved with some level of residual risk to the public.[24] However, the AEA also includes provisions (primarily in Section 161) containing language that allows the USNRC to "*govern...as the Commission may deem necessary or desirable to...protect health or to minimize danger to life*."[30] These provisions: (1) suggest that Congress has granted the USNRC broad discretionary authority to take actions that go beyond ensuring adequate protection of public health and safety in executing its statutory functions; and (2) provide the basis for imposing additional safety enhancements that aim to further reduce the residual risk to the public. Moreover, this view has been further supported by Federal court decisions involving relevant case law in which the courts evaluated both language in AEA provisions and the legislative history to determine Congressional intent in establishing the USNRC's statutory mandate.[24]

The adequacy of public protection from the risks of accidents involving operating reactor units at NPP sites has been debated for many years. The USNRC has historically applied the adequate protection standard in a qualitative manner, relying primarily on established engineering principles and sound technical judgment. In a broad sense, reasonable assurance that the public will be adequately protected from the risks of NPP operations is provided through demonstrating compliance with the collective body of USNRC regulatory requirements that specify conditions that must be met to receive a construction permit and a license to operate an NPP.

The USNRC developed and evaluated safety goals in response to contentious debates among diverse stakeholder groups about the adequacy of NPP safety following the 1979 accident involving Unit 2 at the Three Mile Island Nuclear Station;

while some stakeholders believed additional regulatory requirements should be imposed to further improve NPP safety, others held that the costs of such actions would not be justified in light of what they perceived to be relatively low levels of residual risk from NPP accidents. In its final report, a Presidential commission appointed to investigate the accident urged the USNRC to state its position on this controversial issue.[31] In response, the USNRC declared that it was prepared to move forward with an explicit statement of policy with respect to its safety philosophy and the role of safety-cost tradeoffs in USNRC safety decisions, and thus began its program to develop safety goals.[1] The final safety goal policy statement represents the product of this multi-year effort that included: (1) development of draft policy documents for stakeholder review and comment; (2) public workshops involving participants that represented diverse stakeholder groups;[n] and (3) a two-year trial-use and evaluation period that resulted in some substantive policy changes before final publication and implementation.

The USNRC safety goal policy broadly defines an acceptable level of radiological risk to public health and safety from potential accidental releases of radiological materials from operating power reactors at NPP sites.[o] The safety goal policy

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[n] Examples of stakeholder groups that were represented in the public workshops included: (1) public and environmental interest groups; (2) academic researchers and practitioners from multiple scientific disciplines that had previously contributed to the literature on the topic of acceptable risk, including the social sciences; (3) nuclear utilities and industry advocacy groups; (4) technical experts from the USNRC Advisory Committee on Reactor Safeguards (ACRS); and (5) USNRC technical staff members, managers, and policymakers.
[o] The scope of the USNRC safety goal policy includes risks to the public arising from both routine and accidental radiological releases from operating power reactors at commercial NPPs. The USNRC excluded public risks imposed by the nuclear fuel cycle based on earlier assessments that suggested fuel cycle risks were relatively small in comparison to operating power reactor risks. In addition, environmental impact assessments performed before existing NPPs had been licensed to operate indicated there would be no measurable radiological impact on members of the public from routine operations. Moreover, since compliance with Federal Radiation Council guidance and USNRC regulations was believed to ensure that public risks arising from routine emissions were comparatively small, the USNRC expressed its belief that such risks need not be routinely analyzed on a case-specific basis to

communicates one aspect of the USNRC's risk management philosophy; it reflects a value judgment that essentially addresses the question "*How safe is safe enough?*" for regulatory decisions regarding NPP safety. In practice, it guides agency evaluations to determine whether proposed regulatory actions that would impose additional requirements to enhance NPP safety beyond those needed to ensure adequate protection are justified based on their potential safety benefit, relative to the level of residual risk to the public. Using this approach, proposed regulatory actions that aim to enhance NPP safety can be rejected—even if potentially cost-beneficial—because the level of residual risk to the public is acceptably low, and limited resources could thus be better applied to alternative courses of action. The concept of opportunity cost from microeconomic theory supports this reasoning. In particular, since society has finite resources to expend on safety enhancements, excessive spending to reduce health and safety risks attributed to the nuclear industry could potentially increase net public risk by diverting scarce safety resources from application to potentially more cost-beneficial uses.

The safety goal policy is based on a hierarchical framework comprised of two high-level qualitative safety goals supported by two lower-level QHOs that can be used to determine whether and to what extent each qualitative safety goal has been achieved. The first qualitative safety goal addresses risks to individual members of the public and is supported by the early fatality risk QHO. The second qualitative safety goal is intended to address societal risk and is supported by the latent cancer fatality risk QHO. Figure 5 illustrates this hierarchical framework, while Table I provides the exact language used to specify the qualitative safety goals and

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demonstrate conformance with the safety goal policy.[1] This research therefore focuses only on the application of USNRC safety goals in the context of public risks arising from potential accidental releases of radiological materials from operating power reactors at commercial NPPs.

supporting QHOs in the USNRC Safety Goal Policy Statement.[1]



**USNRC SAFETY GOAL POLICY STATEMENT**

**QUALITATIVE SAFETY GOALS**

**Individual Risk**
Individuals bear no significant additional risk to life and health

**Societal Risk**
Comparable to or less than risks of generating electricity by viable competing technologies

**QUANTITATIVE HEALTH OBJECTIVES**

**Individual Early Fatality Risk**
(0-1.6 km [1 mi])
≤ 0.1% sum from other accidents

**Individual Latent Cancer Fatality Risk**
(0-16.1 km [10 mi])
≤ 0.1% sum from all other causes of cancer

**Figure 5. Hierarchical Framework of the USNRC Safety Goal Policy.** The USNRC safety goal policy is based on a hierarchical framework comprised of two high-level qualitative safety goals supported by two lower-level QHOs that can be used to determine whether and to what extent each qualitative safety goal has been achieved.

The safety goals and QHOs are primarily used in the evaluation of proposed regulatory actions as part of regulatory or backfit analyses. The primary decision analysis technique used to evaluate alternatives in these analyses is cost-benefit analysis. The principal metric calculated in these cost-benefit analyses to support decisionmaking is the net present value of net benefits (net value), which represents the difference between the sum of monetized and discounted benefits and the sum of monetized and discounted costs.[2,3]

However, results from NPP Level 3 PRAs or PCAs can be used to evaluate and screen proposed alternatives based on the magnitude of the estimated safety benefit relative to the level of residual risk to the public, before detailed cost-benefit analyses would be performed. In particular, USNRC regulatory analysis guidelines include

guidance for performing an evaluation of proposed regulatory actions with respect to the USNRC safety goals. This safety goal evaluation is designed to identify when a regulatory requirement should not be imposed generically on NPPs because the residual risk to the public is already acceptably low; it is intended to eliminate some proposed regulatory actions from further consideration, regardless of whether they could be justified on the basis of their net value. This safety goal evaluation can also be used to determine whether a proposed generic safety enhancement backfit that does not meet certain exemption criteria provides a substantial increase in the overall protection of public health and safety to warrant further evaluation of the benefits and costs to determine whether they are justified on their net value basis.[2,3]

USNRC regulatory analysis guidelines include explicit safety goal screening criteria related to: (1) changes in CDF, and (2) conditional containment failure probabilities. These criteria—which are intended to provide a balanced consideration of measures to prevent and mitigate core damage accidents—can be used to evaluate results from Level 1 and Level 2 NPP PRAs to determine conformity with subsidiary safety goal objectives based on CDF and large early release frequency (LERF). Although these guidelines do not include explicit screening criteria related to the early fatality risk and latent cancer fatality risk QHOs, corresponding results from NPP Level 3 PRAs for these metrics can be used to evaluate proposed regulatory actions with respect to the safety goals.[2,3]

For those proposed regulatory actions that pass the safety goal screening evaluation, a detailed cost-benefit analysis is performed to estimate the net value. The principal outputs from a NPP Level 3 PRA or PCA that serve as inputs to the cost-benefit analysis are: (1) averted population dose—which is monetized using a conversion factor that ascribes a monetary value to each unit of population dose that

is averted; and (2) averted economic costs, including offsite property damage.[2,3]

Importantly, during development and evaluation of the safety goal policy, the USNRC decided that the safety goals and QHOs would be applied strictly on a per-reactor-unit basis, even for the multi-unit sites that comprise nearly 60% of the operating U.S. NPP sites; the rationale for this decision was to avoid imposing a regulatory bias against multi-unit sites that may be subject to stricter requirements if the safety goals and QHOs were to be applied on a per-site basis.[7] As a result, accident scenarios involving concurrent radiological releases from multiple co-located operating reactor units have—with few exceptions—traditionally been excluded from NPP PRAs and safety goal evaluations as part of regulatory or backfit analyses for proposed regulatory actions.

This existing gap in the scope and application of the safety goal policy can have important implications. In particular, since multi-unit accident scenarios are excluded from the scope of NPP PRAs and supporting PCAs, the true total accident risk for reactor units that are co-located with other units at multi-unit NPP sites may be underestimated. As a result, safety goal evaluations of proposed regulatory actions that aim to further enhance NPP safety beyond the level provided by adequate protection—and thereby reduce the overall residual risk to public health and safety from NPP accidents—may conclude that these proposed actions would not be justified based on a low level of residual risk to the public. These proposed actions would therefore be rejected before detailed cost-benefit analyses would be performed to determine whether this subset of decision analysis techniques indicate the actions could result in a net benefit to society and thus improve societal welfare.

## II.C. Concurrent Accidents Involving Multiple Operating Reactor Units Co-located at a Shared Site

### II.C.1. Results and Insights from Previous Multi-Unit Probabilistic Risk Assessments: The Seabrook Station Probabilistic Risk Assessment

While most PRAs for multi-unit NPP sites have been performed on a per-reactor-unit basis—assuming all other operating reactor units co-located at a shared NPP site are in a safe and stable condition—some previous NPP PRAs have considered the possibility of concurrent accidents involving multi-unit releases. A notable example is the Seabrook Station Level 3 PRA that was completed in 1983 to address potential emergency planning issues for what was then planned to be a two-unit NPP site.[8]

In the Seabrook study, initiating events were organized into three categories: (1) those that would always impact both units; (2) those that would impact both units only under certain conditions; and (3) those that would impact each unit independently. The frequency of two-unit core damage and radiological release events was estimated by: (1) adjusting the frequency basis for initiating events from events per reactor-year to events per site-year for the two-unit NPP site; (2) developing a simplified logic model that included events involving both units; and (3) using an adaptation of the beta-factor method for treatment of common-cause failure (CCF) events[32] involving SSCs in both units. The conditional consequences of the two-unit radiological release events were estimated by: (1) assuming the two-unit releases would occur simultaneously; and (2) using frequency adjustment factors to adjust either the source term parameters or the conditional consequence estimates obtained from the single-unit PCAs for specific release categories of interest in the two-unit PRA.[8]

Under these assumptions, the Seabrook study demonstrated that—while single-unit accident scenarios provided the greatest contribution to total site risk with respect to both early fatalities and latent cancer fatalities—there were two important findings with respect to two-unit accident scenarios: (1) the two-unit accident scenarios were the dominant contributor to accidents with the greatest consequences in the extreme tails of the CCDF curves; and (2) even for a NPP site with limited sharing of SSCs, two-unit accident scenarios provided a non-negligible contribution of 7% to the total site CDF. For the latter finding, it was further estimated that the conditional probability of occurrence for a two-unit core damage accident, given that a core damage accident had occurred in either unit, was 14%.[8]

Together with findings from other studies that included a limited treatment of multi-unit accident scenarios, these findings suggest that the contribution to total site accident risk from multi-unit accident scenarios is not negligible and could be significant, depending on site-specific factors that influence the potential for dependent failure events or adverse interactions across multiple units.

### II.C.2.  Insights from Reviews of U.S. Operating Experience and LER Data

Under existing USNRC regulations, licensees are required to submit an LER to the USNRC within a specified time period after abnormal conditions[p] are observed at a licensed NPP. These LERs identify and describe the apparent root causes of the abnormal conditions, as well as actions the licensee will take to resolve the issue(s).

Although the USNRC does not specifically record, analyze, and report events involving multiple units at NPP sites, some reports in the USNRC LER database include information about such multi-unit events. A recent study showed that 9%

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[p] Abnormal conditions are those that are beyond the technical specifications that define the conditions under which a nuclear power plant is allowed to operate.

(391 out of 4207) of LERs reported to the USNRC from 2000 through 2011 affected multiple units at a shared NPP site.[12,13] This study also included examples of the types of multi-unit dependencies associated with these events, as well as summaries of the USNRC's evaluation of the significance of selected multi-unit events.

From this research, a classification scheme was developed to characterize potential dependencies across multiple units so that multiple, independent single-unit PRA models could to be integrated into a single multi-unit PRA model for a shared NPP site. Six categories of multi-unit dependencies were identified and developed: (1) initiating events; (2) shared connections; (3) identical components; (4) proximity dependencies; (5) human dependencies; and (6) organizational dependencies.[12,13]

Among other findings, this research demonstrated that adverse events involving multiple operating reactor units co-located at a shared NPP site can and do occur at a non-negligible frequency.

### II.C.3.  The 2011 Accident at the Fukushima Daiichi Nuclear Power Station: A Salient Example of Potential Multi-Unit Accident Scenarios

On March 11, 2011, the Great East Japan (Tohoku) Earthquake—one of the most powerful earthquakes in recorded history—occurred off the northeast coast of Japan. This magnitude 9.0-earthquake caused seafloor deformation that triggered a devastating tsunami that flooded about a 2000-kilometer segment of the Japan coast with inundation heights of up to 40 meters, and injured or killed approximately 25,000 people.[11]

Flooding caused by the earthquake-induced tsunami set in motion a cascade of events that culminated in severe damage to multiple operating reactor units at the FDNPS. At the time of the earthquake, three out of the six FDNPS reactors (Units 1,

2, and 3) were operating at their full rated power level; the remaining three reactors

(Units 4, 5, and 6) were in outages for refueling, maintenance, or inspection

activities. All three operating reactors automatically shut down when the earthquake

occurred. Although the earthquake caused a loss of offsite power (LOOP) initiating

event at the FDNPS, onsite emergency diesel generators (EDGs) initially started and

ran successfully to supply backup power to safety-critical SSCs. However, the

subsequent tsunami eventually flooded EDG intakes, causing the EDGs to fail.[9-11]

This combination of a LOOP caused by the earthquake and subsequent failure of the

EDGs caused by tsunami-induced flooding resulted in a loss of all electrical power to

safety-critical SSCs—a scenario that is commonly referred to as a "*station blackout*"

(SBO). In the subsequent 72 hours, the operating reactors melted down, releasing

hydrogen and radioactive materials into the surrounding containments. Subsequent

hydrogen explosions in the containment buildings caused severe structural damage,

resulting in prolonged releases of very large amounts of radioactive materials into the

surrounding environment.[9-11]

The 2011 Fukushima nuclear accident has been rated as one of the worst nuclear

accidents in history. Along with the 1986 nuclear accident at the Chernobyl Nuclear

Power Station, it is only one of two nuclear accidents to be assigned the worst

possible rating on the International Nuclear and Radiological Event Scale (INES).

The INES scale was developed to facilitate understanding of and communication

about the safety significance of events involving various sources of radiation,

including NPPs. Safety significant events involving nuclear or radiological materials

are classified on the INES scale at one of seven levels: Levels 1-3 are categorized

as "*incidents*" and Levels 4-7 are categorized as "*accidents.*" Events without safety

significance are considered to be below the INES scale and are assigned a rating of

Level 0.[33]

Using the INES scale, events are classified based on their impact on three broad areas: (1) people and the environment; (2) radiological barriers and controls at affected facilities; and (3) defense-in-depth considerations.[33] Similar to the Richter scale for earthquakes, the INES scale is logarithmic; each increase in INES level is intended to represent an order of magnitude increase in the severity of the event consequences.

The Fukushima nuclear accident was assigned the highest possible rating of Level 7, which is a "*major accident*" characterized by a "*major release of radioactive material with widespread health and environmental effects requiring implementation of planned and extended countermeasures*."[11,33] Prior to Fukushima, the Chernobyl nuclear accident was the only nuclear accident to receive an INES rating of Level 7. [33] By comparison, the 1979 accident at Three Mile Island Nuclear Station Unit 2— which resulted in severe damage to the reactor core, but only a minor release of radioactive material beyond its containment structure—was assigned an INES rating of Level 5, which is characterized as an "*accident with wider consequences*."[33]

The 2011 accident at the FDNPS provides a salient example of potential multi-unit accident scenarios. It challenged many assumptions that had been used in previous PRA and PCA studies and underscored the importance of many factors affecting the initiation, progression, and consequences of potential multi-unit accident scenarios; examples include: (1) dependencies or interactions across multiple co-located units at a shared NPP site—especially those influenced by large-scale external events; (2) differences in accident progression timelines across multiple units, resulting in staggered and/or prolonged multi-unit releases; (3) variability in local weather

conditions over time and the need to account for the effects of changes in wind direction on atmospheric transport and dispersion; and (4) while not addressed as part of this research, the potential for multi-unit accident scenarios that include spent fuel pool units co-located with reactor units at a shared NPP site.

## II.D. The State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

### II.D.1. Project Overview

In 2005—six years before the occurrence of the 2011 nuclear accident at the FDNPS—the USNRC initiated the SOARCA project to: (1) develop integrated state-of-the-art reactor accident progression and offsite radiological consequence models that leveraged the enhanced state of knowledge about severe accident phenomena and radiological health effects developed over decades of research, and modeled improvements in NPP design and operation that had not been reflected in earlier studies; and (2) obtain realistic estimates of the public health outcomes for select single-unit accident scenarios that were judged to be important based on their contribution to CDF or their potential to cause offsite radiological health consequences.[14-16]

This research builds upon the state-of-the-art offsite radiological consequence models that were developed as part of the initial pilot study for the SOARCA project, and therefore relies heavily on its underlying technical basis for model and parameter value selection. This section will therefore describe in some detail the objectives, study design, consideration of multi-unit events, and conclusions for the SOARCA pilot study; follow-on SOARCA studies are also briefly described.

### II.D.2.  Project Objectives

The overall objective of the SOARCA project was to develop an updated body of knowledge regarding the realistic outcomes for important severe reactor accidents. This overall objective was complemented by a number of supporting objectives; two of these that directly pertain to this research include:[14-16]

1. **Incorporate integrated modeling of severe accident progression and offsite consequences using state-of-the-art analytical tools.** Leverage the enhanced state of knowledge about severe accident phenomena and radiological health effects that had been developed over the course of several decades of research.

2. **Model modifications to NPP design and operation that were not reflected in previous PRA or PCA studies.** Examples of these changes included: (1) system design enhancements; (2) improved training, emergency operating procedures (EOPs), and ERPs; and (3) Extensive Damage Mitigation Guideline (EDMG) measures implemented following the terrorist attacks on September 11, 2001 to improve each NPP's capability to mitigate events involving loss of large areas caused by fires or explosions.[34]

### II.D.3.  Pilot Study Design

Two representative NPP sites using the traditional large light-water reactor (LWR) design were selected for an initial pilot study. The NPP sites evaluated in the SOARCA pilot study were: (1) Peach Bottom, Unit 2—located approximately 18 miles south of Lancaster, PA; and (2) Surry, Unit 1—located approximately 17 miles northwest of Newport News, VA. Peach Bottom is generally representative of U.S. operating reactors using the General Electric boiling-water reactor (BWR) design with a Mark I containment. Surry is generally representative of U.S. operating

reactors using the Westinghouse pressurized-water reactor (PWR) design with a large, dry containment.

Accident scenarios were selected for detailed modeling and evaluation using a rigorous process that coupled results and insights from available PRA models for each site with expert judgments about the relative importance of each scenario. To focus study resources, criteria were developed to identify the most important accident scenarios based on two factors: (1) their likelihood of causing core damage—which was assessed using their contribution to CDF; and (2) their potential for causing significant offsite radiological health consequences due to an early failure or bypass of the containment structure. In particular, accident scenarios were selected for inclusion in the SOARCA pilot study if: (1) their CDF contribution was equal to or greater than 1E-06 per reactor-year; or (2) they involved early failure or bypass of containment and their CDF contribution was equal to or greater than 1E-07 per reactor-year.[14-16] Using this approach, seven single-unit accident scenarios were selected for detailed modeling and evaluation under the SOARCA pilot study. These accident scenarios are summarized in Tables II and III.

To assess the potential benefits of EDMGs and to provide a basis for comparison to the previous analyses of unmitigated severe accident scenarios, the SOARCA pilot study analyzed each scenario with and without crediting EDMG equipment and procedures. The analysis that credits successful implementation of the mitigation measures—in addition to actions directed by the EOPs and SAMGs—is referred to as the "*mitigated case*". The analysis that does not credit these mitigation measures is referred to as the "*unmitigated case*."[14-16] Since a formal human reliability analysis (HRA) was not performed to quantify human error probabilities (HEPs) for the modeled operator actions in the mitigated cases, only the unmitigated cases are

used for the purposes of this research.

### II.D.4.  Consideration of Multi-Unit Events

Both Peach Bottom and Surry are multi-unit NPP sites, each comprised of two operating reactor units at a shared site. During the accident scenario identification and selection process described above, analysts identified potential scenarios in which both units at each NPP site could concurrently experience initiating events and subsequent event sequences that lead to concurrent core damage.

Although the CDF contribution from these multi-unit accident scenarios was in the range of the 1E-06 per reactor-year inclusion criterion, treatment of these scenarios was determined to be beyond the scope of the SOARCA pilot study. Instead, the issue of potential multi-unit core damage accident scenarios was proposed as a safety-related generic issue and referred for further evaluation because it was recognized that such accident scenarios may: (1) challenge the ability of the NPP operating personnel to respond and may require resources beyond those that are available for single-unit accident scenarios; and (2) increase the amount of radiological material released to the environment and the subsequent offsite radiological consequences. However, a screening panel later recommended that the issue not be treated as part of the Generic Issues Program. This recommendation was based on: (1) the results of a scoping analysis that aimed to develop a bounding estimate of multi-unit risk using results from single-unit PRAs—which suggested that the issue was of low risk significance to public health and safety; and (2) the need for longer term efforts to develop analytical tools that could more accurately estimate the risks attributed to multi-unit accident scenarios.[35]

### II.D.5.  Pilot Study Conclusions

Based on the results from the integrated state-of-the-art accident progression and offsite radiological consequence models for Peach Bottom and Surry, the SOARCA pilot study yielded a number of key conclusions:[14-16]

1.  When operators are successful in using available onsite equipment during the modeled accident scenarios, they can prevent reactor core damage or delay or reduce radiological releases to the environment.

2.  For all modeled accident scenarios—whether mitigated or unmitigated— accidents progress more slowly and release much smaller amounts of radiological material than estimated in previous PRA and PCA studies.

3.  Delays in estimated radiological releases provide more time for implementing offsite emergency response actions. If ERPs are implemented as planned and practiced, offsite emergency response actions reduce the risk of radiological health consequences attributed to modeled accident scenarios.

4.  For all modeled severe accident scenarios—whether mitigated or unmitigated— there is essentially no risk of early fatalities during or shortly after the accident.

5.  Latent cancer fatality risks attributed to the modeled accident scenarios are millions of times lower than the background cancer fatality risk for the general U.S. population.

Although these findings are based on a limited set of accident scenarios that were analyzed using site-specific models for the Peach Bottom and Surry NPP sites, they may be generally applicable to NPPs with similar designs. Additional studies described in the next two subsections are being performed to expand upon the

insights derived from the SOARCA pilot study and to assess their applicability to other reactor-containment designs and NPP sites.

### II.D.6.  SOARCA Uncertainty Analysis

Uncertainty analyses are being performed for specific accident scenarios that were evaluated as part of the SOARCA pilot study. These analyses are conditioned on the assumed occurrence of specified conditions in the progression of the modeled accident scenarios. There are three high-level objectives for these conditional uncertainty analyses: (1) develop insights into the overall sensitivity of SOARCA results to uncertainty in inputs; (2) identify the most influential input parameters for accidental radiological releases and accident consequences; and (3) demonstrate the application of an uncertainty analysis methodology that could be used in future source term, PCA, or NPP Level 3 PRA studies.[36]

The uncertainty analyses involve varying multiple uncertain model parameters using Monte Carlo sampling of parameter probability distributions. Subject matter experts were consulted to determine the most important uncertain parameters in accident progression, radiological release, and offsite radiological consequence models for variation. Multiple statistical regression techniques are then used to quantify uncertainty and to determine which parameters have the greatest influence on the results.[36]

### II.D.7.  SOARCA Study for the Sequoyah Nuclear Plant

A follow-on SOARCA study was initiated to develop best estimates of the offsite radiological health consequences for select accident scenarios involving the Sequoyah Nuclear Plant—which uses a PWR design with an ice condenser containment. Accident scenarios were chosen to specifically challenge this style of

containment, which is smaller than the large, dry containment used with other PWR

NPPs, including Surry.[37]

The Sequoyah SOARCA study is applying modeling lessons learned and best

practices from the SOARCA pilot study.[38,39] In addition, the effects of using diverse

and flexible coping strategies involving portable equipment—which the U.S. nuclear

industry implemented in response to challenges identified by the 2011 accident at

the FDNPS[40]—are also being modeled as part of this study to characterize their

benefits.

# Chapter III: Methods and Analytical Tools

## III.A.   Methods for Estimating Contribution to Safety Goal QHO Risk Metrics from Modeled Single-Unit and Two-Unit Accident Scenarios

### III.A.1. Overview

The safety goal QHOs were developed for comparison with safety goal QHO risk metric results from full-scope NPP Level 3 PRAs that model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential accident scenarios. The SOARCA pilot study[14-16] used state-of-the-art methods, models, data, and analytical tools to develop more realistic estimates of the offsite radiological consequences attributed to nuclear accident scenarios. To leverage the advanced models, results, and insights that were developed as part of this effort, this contemporary study was selected as the foundation to build upon for this research. However, the SOARCA pilot study was not a full-scope NPP Level 3 PRA; it was a limited-scope PCA study that performed detailed modeling and integrated analysis of accident progression and offsite consequences for a small set of single-unit accident scenarios that were judged to be important.

To perform an adequate evaluation of the effect of expanding the scope of the safety goal policy to include the contribution from multi-unit accident scenarios to safety goal QHO risk metrics thus required development of novel methods for: (1) converting the single-unit accident scenario frequencies and consequences from the SOARCA pilot study to representative single-unit accident risk results for each NPP site; (2) estimating frequencies for all modeled two-unit accident scenarios; (3) implementing the new multi-source model in MACCS to estimate the conditional consequence contribution to safety goal QHO risk metrics for all modeled two-unit

accident scenarios under base case and sensitivity analysis assumptions; (4)

converting the two-unit accident scenario frequencies and conditional consequences

estimated as part of this research to representative two-unit accident risk results; and

(5) estimating the figures of merit used to evaluate the effect of this expansion in

scope of the safety goal policy.

Figures 6, 7, and 8 respectively illustrate the process used to implement a novel

approach that was developed for estimating the contributions to safety goal QHO risk

metrics from single-unit accident scenarios, from multi-unit accident scenarios, and

the figures of merit. More detailed descriptions about the process steps, equations,

variables, and data sources that correspond to each figure are provided in the

following subsections. Although these figures and descriptions focus on the case

study application of this process to representative two-unit NPP sites, some ideas

regarding how this approach could be generalized and applied to NPP sites with

differing numbers of units are provided in Section III.C.

*III.A.2. Estimation of the Contribution from Single-Unit Accidents to Safety*
*Goal QHO Risk Metrics (Figure 6)*

**Step 1, Step 2a, and Step 2b.** These steps respectively involve: (1) selecting and

modeling single-unit accident scenarios judged to be important to risk; (2a)

estimating their corresponding frequencies; and (2b) estimating their conditional

consequences—in this case, conditional consequence metrics that provide input to

the risk metrics for comparison to the early fatality risk and latent cancer fatality risk

QHOs. For this research, all of these steps were accomplished as part of the

SOARCA pilot study that provides much of the underlying technical basis for the

state-of-the-art consequence models used in this research. [14-16]
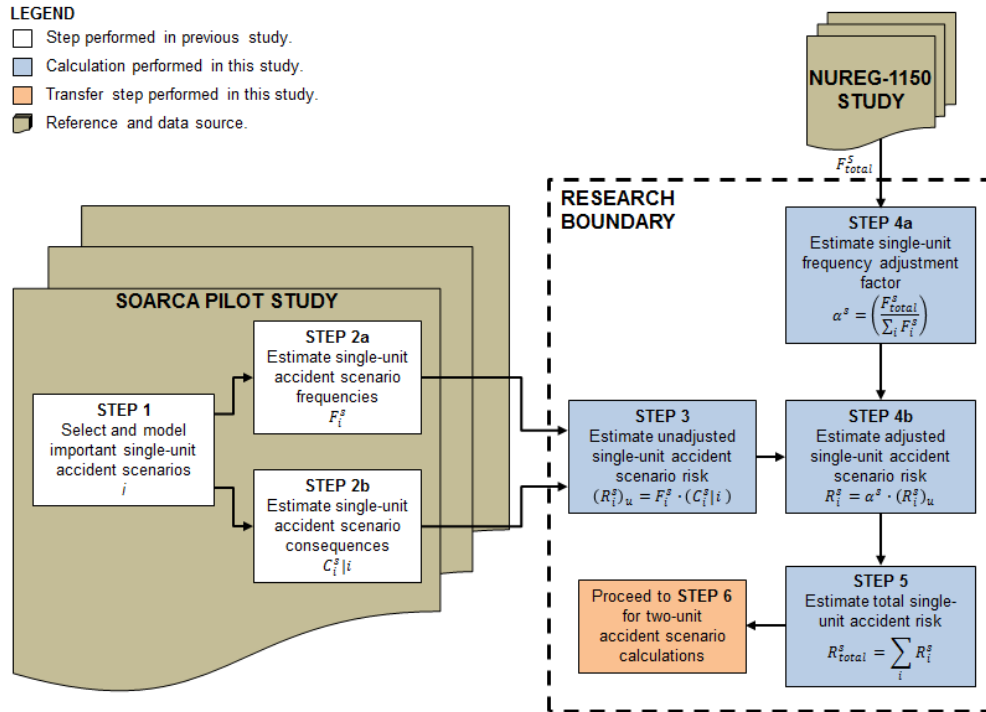
**Figure 6. Process for Estimating the Contribution from Single-Unit Accidents to Safety Goal QHO Risk Metrics.** A two-unit case is used to demonstrate the process for estimating the contribution from single-unit accident scenarios to safety goal QHO risk metrics.
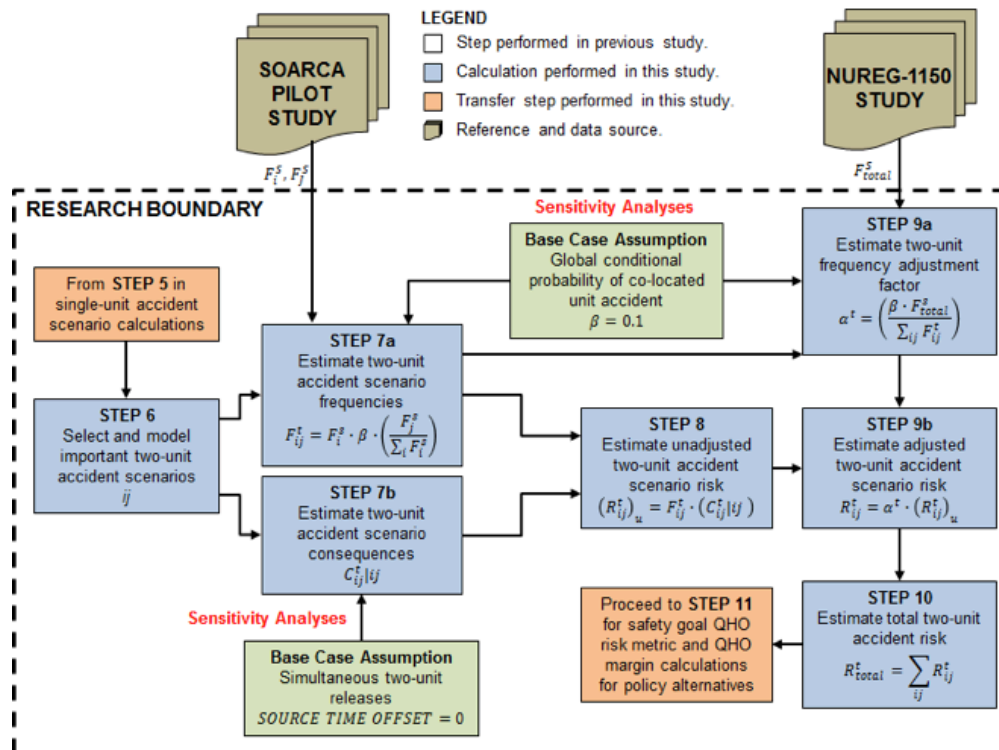


**Figure 7. Process for Estimating the Contribution from Multi-Unit Accidents to Safety Goal QHO Risk Metrics.** A two-unit case is used to demonstrate the process for estimating the contribution from multi-unit accident scenarios to safety goal QHO risk metrics.

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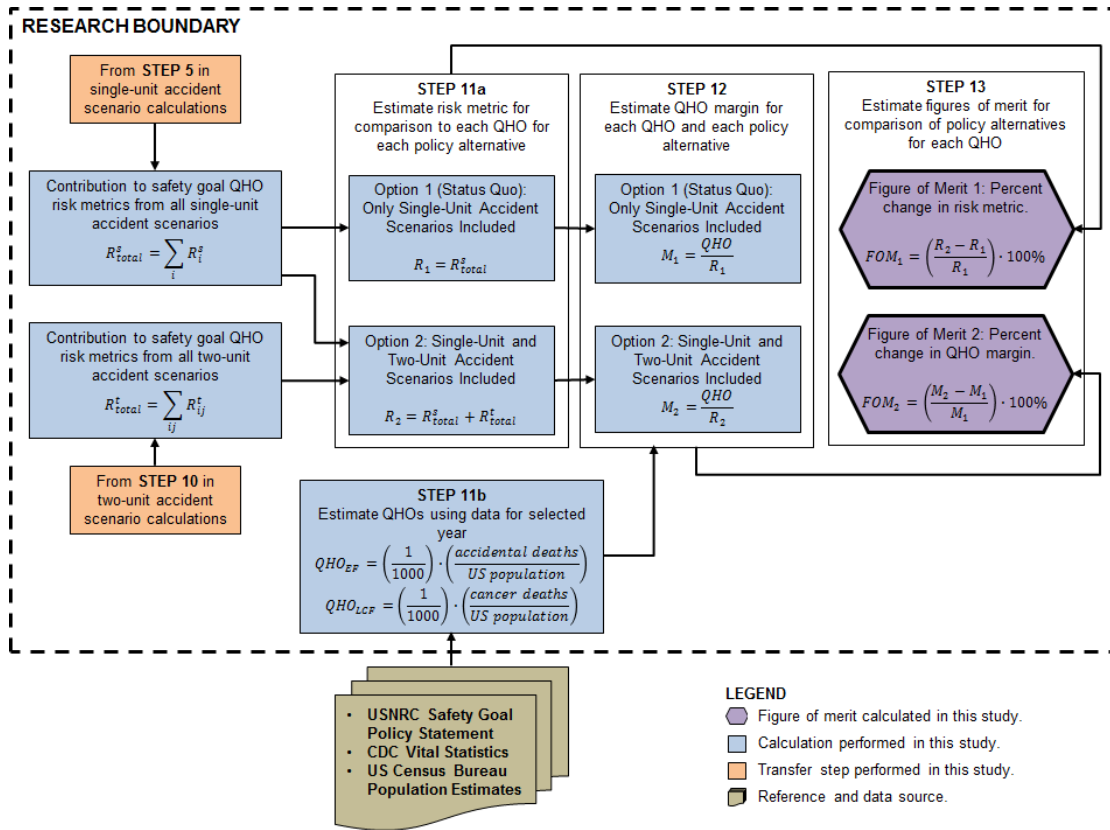
**Figure 8. Process for Estimating Figures of Merit for Evaluation of Policy Alternatives.** A two-unit case is used to demonstrate the process for estimating risk metrics for comparison to the safety goal QHOs and the figures of merit for evaluation of policy alternatives.

**Step 3: Estimate unadjusted single-unit accident scenario risk.** The unadjusted contribution to risk from each single-unit accident scenario $i$ is estimated using Equation (1):

$$(R_i^s)_u = F_i^s \cdot \left(C_i^s | i \right) \qquad (1)$$

- $(R_i^s)_u$ = unadjusted mean risk contribution from single-unit accident scenario $i$ to each safety goal QHO risk metric. Unadjusted means the risk contribution has not been adjusted to account for the contribution to frequency from other single-unit accident scenarios in a similar class that scenario $i$ is assumed to represent.

- $F_i^s$ = mean single-unit CDF contribution from single-unit accident scenario $i$.

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Estimates for the Peach Bottom and Surry single-unit accident scenario mean CDF contributions were provided in the SOARCA pilot study.[14-16]

- $(C_i^S|i)$ = mean value for the single-unit conditional safety goal QHO consequence metrics, conditioned on the occurrence of representative single-unit accident scenario *i*. This represents the consequence contribution to the safety goal QHO risk metrics from the class of single-unit accidents that scenario *i* is assumed to represent. Estimates of the mean conditional consequences for the Peach Bottom and Surry single-unit accident scenarios were calculated in the SOARCA pilot study using the MACCS code.[14-16]

**Step 4a: Estimate single-unit frequency adjustment factor.** An adjustment factor is needed to adjust the single-unit accident scenario risk estimates to account for the contribution to the frequency element of the risk triplet from other single-unit accident scenarios in each class that have not been modeled and analyzed. This approach assumes that each single-unit accident scenario *i* results in conditional consequence distributions that are similar to those that would result from each of the other single-unit accident scenarios that scenario *i* is assumed to represent. Moreover, it is assumed that a global adjustment factor can be applied to all single-unit accident scenarios; this implies that the proportion of single-unit accident scenarios that are not modeled is the same for all classes of single-unit accident scenarios that are represented. The single-unit frequency adjustment factor can be estimated using Equation (2):

$$\alpha^S = \left(\frac{F_{total}^S}{\Sigma_i F_i^S}\right) \qquad (2)$$

- $\alpha^s$ = global single-unit frequency adjustment factor.

- $F_{total}^s$ = mean total single-unit CDF from all single-unit accident scenarios initiated by internal events, fires, and seismic events. Estimates of the single-unit CDFs from all three initiating event hazards for Peach Bottom and Surry were provided in the NUREG-1150 study[41] that documented NPP Level 3 PRAs for five U.S. NPP sites. Although these NPP Level 3 PRA studies were completed more than 20 years ago, the CDF estimates are still considered among the NPP PRA community to be sufficient for analyses such as this for which an order-of-magnitude estimate of CDF is considered to be sufficient. Moreover, since the figures of merit for this evaluation represent relative changes to isolate the effect of including the risk contribution from multi-unit accidents, it can be mathematically demonstrated that this variable does not influence these figures of merit; it only influences that absolute values of the safety goal QHO risk metrics and QHO margin.

- $\sum_i F_i^s$ = combined frequency of all modeled single-unit accident scenarios that are assumed to be representative of the full spectrum of potential single-unit accident scenarios.

**Step 4b: Estimate adjusted single-unit accident scenario risk.** The adjusted contribution to risk from each single-unit accident scenario *i* is estimated using Equation (3):

$$R_i^s = \alpha^s \cdot (R_i^s)_u \qquad (3)$$

- $R_i^s$ = adjusted mean risk contribution to each safety goal QHO risk metric from the class of single-unit accidents that scenario *i* is assumed to represent.

Adjusted means the risk contribution has been adjusted to account for the contribution to frequency from other single-unit accident scenarios in the same class that are not modeled.

**Step 5: Estimate total single-unit accident risk.** The total contribution to risk from all single-unit accident scenarios is estimated using Equation (4):

$$R^s_{total} = \sum_i R^s_i \qquad (4)$$

- $R^s_{total}$ = total mean risk contribution to each safety goal QHO metric from all single-unit accident scenarios initiated by internal events, fires, and seismic events.

*III.A.3. Estimation of the Contribution from Two-Unit Accidents to Safety Goal QHO Risk Metrics (Figure 7)*

**Step 6: Select and model important two-unit accident scenarios.** The process used to select the two-unit accident scenarios for modeling and analysis as part of this research was described in Section I.E.3. A total of 25 two-unit accident scenarios (nine for Peach Bottom and 16 for Surry) were constructed using all possible combinations of single-unit accident scenarios that were modeled for each NPP site as part of the SOARCA pilot study.

A description about the process used to model each two-unit accident scenario first requires a description about: (1) the MelMACCS interface software utility; (2) the new multi-source modeling capability that has been implemented in MACCS; and (3) identification of the additional parameters that must be defined to develop a MACCS model for a multi-unit accident scenario, relative to developing a MACCS model for a

single-unit accident scenario. The technical basis that underlies the

phenomenological models used in MACCS, as well as information about the

verification and validation testing that it has undergone, are well documented in

accessible resources.[17,18]

## **MelMACCS and the MACCS Multi-Source Model**

The USNRC uses the MELCOR code for modeling severe accident progression and

estimation of source terms.[38] Source term information is contained in MELCOR plot

files (*.ptf) that are processed using the MelMACCS interface software utility to

extract the data needed for the offsite radiological consequence analysis and to

generate the corresponding inputs for the MACCS consequence model.

MACCS was recently enhanced in Version 3.10[q] to include the capability to model

releases from multiple, co-located radiological sources with potentially different

accident progression timelines and unique radionuclide inventories. Figure 9

provides an overall conceptual model illustrating the relationships between key

inputs and phenomena modeled in MACCS to calculate conditional consequence

metrics for two-unit accident scenarios.

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[q] MACCS Version 3.10 was used to perform all MACCS modeling and simulations in support of this research.
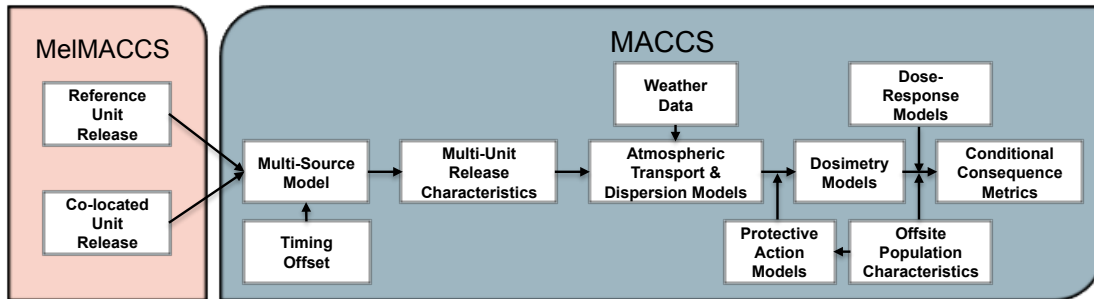
**Figure 9. Overall Conceptual Model.** A conceptual model illustrates relationships between key inputs and phenomena modeled in MACCS to calculate conditional consequence metrics for two-unit accident scenarios.

The new multi-source model is implemented by assigning values to an additional set of parameters that completely specify the multi-source model. Table IV summarizes these parameters and provides an explanation for how each parameter was treated for this research. In practice, once the analyst identifies the MelMACCS-generated source term input files that are to be combined within the multi-source model, MACCS calculates the values for most parameters based on the number of source term input files that are specified and the information they contain with respect to the numbers of plume segments and delay times.

**Modeling of Two-Unit Accident Scenarios**

For this research, the only parameter that required a decision about its value was the SOURCE TIME OFFSET parameter that defines the timing offset for each of the source term input files. Moreover, only the timing offset for the accident in the co-located unit needed to be defined to fully specify the multi-source model for all two-accident scenarios. There were two reasons for this: (1) this research assumes that one of the single-unit accident scenarios is always considered to be the reference point with respect to the timing of accident initiation, progression, and radiological

releases—which means the value of SOURCE TIME OFFSET for the reference unit accident scenario is always set to 0; and (2) Peach Bottom and Surry are both comprised of two operating reactor units—which means the number of source term input files for each multi-unit accident scenario is always two, since accidents involving spent fuel pool units and dry cask storage facilities are excluded from this research (NUM_SOURCES = 2).

To evaluate the effect of variation in the SOURCE TIME OFFSET parameter on the impact of including the contribution from multi-unit accident scenarios to the safety goal QHO risk metrics, the following analyses were performed with alternative values assigned to the SOURCE TIME OFFSET parameter for the co-located unit:

1. **Base Case Analysis:** The value of SOURCE TIME OFFSET was set to 0 to model simultaneous concurrent accident scenarios in both the reference unit and its co-located unit.

2. **One-Way Sensitivity Analyses:** The value of SOURCE TIME OFFSET was varied from 1 to 7 days in one-day increments to model and evaluate the effects of differences in the timing of accident initiation, progression, and radiological releases for the reference unit and co-located unit. This range of plausible values was selected because it was judged to provide sufficient results to determine whether any patterns or trends emerge.[r]

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[r] A limited set of sensitivity analyses were also performed to evaluate the effect of varying the timing offset parameter from 1 to 24 hours in one-hour increments. Only results for one-way sensitivity analyses using the 1 to 7 day range in one-day increments are presented for two reasons: (1) results did not converge within 24 hours for some two-unit accident scenarios, indicating the need to perform sensitivity analyses using timing offsets greater than 24 hours; and (2) one-day increments appeared to provide sufficient resolution to characterize patterns or trends.

**Table IV. Parameters Unique to the MACCS Multi-Source Model**

| Parameter | Description | Range of Parameter Values | | Units | Treatment |
|-----------|-------------|------------|-------------|-------|-----------|
| | | Lower Limit | Upper Limit | | |
| TOTREL | Defines number of plume segments released over all files specified for multi-source model. | 2 | 500 | N/A | Maintained number of plume segments defined in each of the SOARCA study source term files. Multi-source calculates TOTREL by summing the number of plume segments defined in each source term input file. |
| MS_LABELS | MelMACCS file name for source term input files. | 1-255 characters | | N/A | Defined when multi-source term file set specified in MACCS file specifications. |
| PLUME_DLY | Start time of plume release relative to MELCOR time frame. | 0 | 2592000 | seconds | Maintained plume delay timing defined each of the SOARCA study source term files. |
| NUM_SOURCES | Number of source term files specified. Defined when user specifies multi-source term file set. | 2 | 500 | N/A | Defined when multi-source term file set is specified in MACCS file specifications. Since both Peach Bottom and Surry are two-unit sites, this study was limited to modeling and analyzing multi-source accident scenarios involving exactly two source terms. (NUM_SOURCES=2) |
| SOURCE TIME OFFSET | Timing offset for each specified source term file. | 0 | 2592000 | seconds | Time offset for first source term file was set to 0 to coincide with modeled reference unit accident initiation. Time offset for co-located unit source term file was set to 0 for the base case assumption of simultaneous accident scenarios and varied from 1 to 7 days in one-day increments for sensitivity analyses to evaluate effects of variation in timing offet for concurrent accident scenarios. |

**Step 7a: Estimate two-unit accident scenario frequencies.** Estimation of two-unit accident scenario frequencies requires consideration of three elements: (1) the unconditional frequency of the single-unit accident scenario initiated in the reference unit ($F_i^S$); (2) the conditional probability of an accident occurring in the co-located unit, given the reference unit's accident scenario frequency; and (3) the conditional probability of the co-located unit experiencing accident scenario *j* concurrent with the reference unit's accident scenario, given that single-unit accident scenario *i* has occurred in the reference unit and given that a concurrent accident has occurred in the co-located unit.

Two alternative approaches to estimating the last element were considered. The first approach was to assign an equal conditional probability to each of the single-unit

accident scenarios in the co-located unit that the reference unit accident scenario

could be combined with. However, this approach was discarded in favor of an

approach that utilizes important information about the unconditional relative

likelihoods of each accident scenario. This was judged to be appropriate because

whether or not a specific accident scenario has occurred in the reference unit, there

are certain attributes that still make some accident scenarios in the co-located unit

less likely to occur than others. For example, an ISLOCA scenario for Surry Unit 2

should still be less likely than the LTSBO scenario, regardless of which single-unit

accident scenario Surry Unit 1 experienced. Based on the adopted approach, the

frequency of each two-unit accident scenario is estimated using Equation (5):

$$F_{ij}^t = F_i^s \cdot \beta \cdot \left( \frac{F_j^s}{\Sigma_i F_i^s} \right) \qquad\qquad (5)$$

- $F_{ij}^t$ = mean two-unit CDF contribution from two-unit accident scenario *ij*. The

  index *i* is used to represent the single-unit accident scenario that occurs in the

  reference unit, while the index *j* is used to represent the single-unit accident

  scenario that concurrently occurs in the co-located unit.

- $\beta$ = global conditional probability of an accident occurring in the co-located unit,

  given that single-unit accidents involving the reference unit occur at a specified

  frequency. For this research, a global average conditional probability is assumed

  to apply across all multi-unit accident scenarios. In reality, each multi-unit

  accident scenario can have a unique conditional probability given the occurrence

  of a specified single-unit accident scenario in the reference unit. For the base

  case analysis, the value of $\beta$ was assumed to be 0.1; this implies a 10% chance

  of the co-located unit experiencing a core damage accident, given that core

  damage accidents occur in the reference unit at the estimated frequency. Based

on results and insights from previous multi-unit NPP PRA studies[8] and operating experience data,[12,13] this appeared to be a reasonable assumption. One-way sensitivity analyses were then performed to evaluate the effect of varying this parameter over its entire range of possible values from 0 to 1.

- $\left(\frac{F_j^S}{\sum_i F_i^S}\right)$ = conditional probability of single-unit accident scenario $j$ concurrently occurring in the co-located unit, given that single-unit accident scenario $i$ has occurred in the reference unit and that a concurrent accident has occurred in the co-located unit. This represents the relative contribution of each single-unit accident scenario frequency to the combined single-unit accident frequency from the set of all modeled single-unit accident scenarios for each NPP site.

**Step 7b: Estimate two-unit accident scenario consequences.** The mean values for the two-unit conditional consequence contribution to the safety goal QHO risk metrics, conditioned on the occurrence of representative two-unit accident scenario $ij$ $\left(C_{ij}^t \middle| ij\right)$, are estimated by implementing the multi-source model in the MACCS code. This represents the conditional consequence contribution to the safety goal QHO risk metrics from the class of two-unit accidents that two-unit accident scenario $ij$ represents.

**MACCS Output Options for Estimating Conditional Consequence Contribution to Safety Goal QHO Risk Metrics**

MACCS has the ability to calculate and report two different consequence metrics that can be used to estimate the conditional consequence contribution to the safety goal QHO risk metrics for each modeled accident scenario:

1.  **Average Individual Health Effect Risk:** For this option, the average individual risk of each specified health effect (i.e. early fatality and latent cancer fatality) is calculated by summing the risk values for all compass sectors in the modeled spatial grid at a given distance and dividing it by the number of compass sectors. This averaging approach effectively assumes there is a uniform population distribution around the NPP site at the specified radial distance from the site.

    While this may not be a realistic assumption, this computational approach is needed to estimate the average individual early fatality risk within one mile of the NPP site boundary for comparison with the corresponding QHO when no individuals reside within one mile of the NPP site boundary. This MACCS output was therefore selected with a radial distance of one-mile for calculating the conditional consequence contribution to the safety goal QHO risk metric for comparison to the early fatality risk QHO for two reasons: (1) no individuals reside within one mile of the Surry site boundary; and (2) although some individuals reside within one mile of the Peach Bottom site boundary, this output metric was also used for Peach Bottom to enable a more meaningful comparison of results and insights with respect to this QHO across reactor and containment designs.[s]

2.  **Population-Weighted Health Effect Risk.** For this option, the population-weighted health effect risk is estimated by calculating the total number of cases of each specified health effect in a defined spatial interval around the NPP site,

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[s] Since average individual early fatality risk at one-mile from the NPP site boundary was selected as the MACCS output to calculate the conditional consequence contribution to the safety goal QHO risk metric for comparison to the early fatality risk QHO, the consequence models developed for the single-unit accident scenarios modeled as part of the SOARCA pilot study had to be modified and run again to produce this specific output. MACCS Version 3.10 was also used to perform these simulations.

and then dividing by the total population in the spatial interval. This output metric

is typically used instead of the average individual health effect risk metric

because its computational approach accounts for the population distribution

around the NPP site, and can therefore produce a more realistic estimate. This

output metric was therefore selected with a defined spatial interval of 0-10 miles

for calculating the conditional consequence contribution to the safety goal QHO

risk metric for comparison to the latent cancer fatality risk QHO.

**Step 8: Estimate unadjusted two-unit accident scenario risk.** The unadjusted

contribution to risk from each two-unit accident scenario *ij* is estimated using

Equation (6):

$$\left(R_{ij}^{t}\right)_{u} = F_{ij}^{t} \cdot \left(C_{ij}^{t}|ij\right) \qquad (6)$$

- $\left(R_{ij}^{t}\right)_{u}$ = unadjusted mean risk contribution to each safety goal QHO risk metric

  from two-unit accident scenario *ij*. Unadjusted means the risk contribution has not

  been adjusted to account for the contribution to frequency from other two-unit

  accident scenarios in a similar class that scenario *ij* is assumed to represent.

**Step 9a: Estimate two-unit frequency adjustment factor.** Similar to single-unit

accident scenarios, an adjustment factor is needed to adjust the two-unit accident

scenario risk estimate to account for the contribution to the frequency element of the

risk triplet from other two-unit accident scenarios that have not been modeled and

analyzed. This approach assumes: (1) conditional consequences from the set of all

modeled two-unit accident scenarios collectively represent the spectrum of

consequence contributions to the safety goal QHO risk metrics from all potential two-

unit accident scenarios; and (2) each two-unit accident scenario *ij* results in

conditional consequence contributions that are similar to those that would result from each of the other two-unit accident scenarios in the class that two-unit accident scenario *ij* is assumed to represent. Moreover, it is assumed that a global adjustment factor can be applied to all two-unit accident scenarios; this implies that the proportion of two-unit accident scenarios that are not modeled is the same for all classes of two-unit accident scenarios that are represented. The two-unit frequency adjustment factor can be estimated using Equation (7)[t]:

$$\alpha^t = \left( \frac{\beta \cdot F_{total}^S}{\sum_{ij} F_{ij}^t} \right) \qquad (7)$$

- $\alpha^t$ = global two-unit frequency adjustment factor.

- $\beta \cdot F_{total}^S$ = mean total two-unit CDF from all two-unit accident scenarios initiated by internal events, fires, and seismic events. This is estimated by the product of the unconditional mean total single-unit CDF estimate obtained from the NUREG-1150 study ($F_{total}^S$) and the global conditional probability of an accident in the co-located unit accident, given a specified reference unit accident frequency ($\beta$).

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[t] Two key assumptions in this research are: (1) the two operating reactor units co-located at each NPP site are identical; and (2) one unit always serves as the reference unit with respect to the timing of accident initiation, progression, and radiological release for two-unit accident scenarios. Under these assumptions, only the 25 two-unit accident scenario consequence models needed to be run for the base case and sensitivity analyses to estimate conditional consequence contributions to safety goal QHO metrics for each two-unit accident scenario. However, since either unit on the site could serve as the reference unit, the frequency for each two-unit accident scenario should—in principle—be doubled to account for the contribution from accidents in which the other unit serves as the reference unit. Since a factor of two would be applied to both the numerator and denominator of Equation (7) to account for this frequency contribution, the value of the two-unit frequency adjustment factor would not be affected. Therefore, while these factors of two should conceptually be included in Equation (7), they are not displayed since reducing the equation to its simplest form would eliminate them.

- $\sum_{ij} F_{ij}^t$ = combined frequency of all modeled two-unit accident scenarios that are assumed to be representative of the full spectrum of potential two-unit accident scenarios.

**Step 9b: Estimate adjusted two-unit accident scenario risk.** The adjusted contribution to risk from each two-unit accident scenario *ij* is estimated using Equation (8):

$$R_{ij}^t = \alpha^t \cdot \left(R_{ij}^t\right)_u \qquad (8)$$

- $R_{ij}^t$ = adjusted mean risk contribution to each safety goal QHO risk metric from the class of two-unit accidents that scenario *ij* is assumed to represent. Adjusted means the risk contribution has been adjusted to account for the contribution to frequency from other two-unit accident scenarios in the same class that are not modeled.

**Step 10: Estimate total two-unit accident risk.** The total contribution to risk from all two-unit accident scenarios is estimated using Equation (9):

$$R_{total}^t = \sum_{ij} R_{ij}^t \qquad (9)$$

- $R_{total}^t$ = total mean risk contribution to each safety goal QHO risk metric from all two-unit accident scenarios initiated by internal events, fires, and seismic events.

## III.B. Methods for Estimating Figures of Merit to Evaluate Policy Alternatives (Figure 8)

**Step 11a: Estimate risk metrics for comparison to each QHO for each safety goal policy alternative.** The safety goal QHO risk metrics[u] that are to be compared with the corresponding QHOs are calculated for each policy alternative evaluated in this research using Equations (10) and (11):

### Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included ($R_1$)

$$R_1 = R_{total}^s \qquad (10)$$

### Option 2: Single-Unit and Two-Unit Accident Scenarios Included ($R_2$)

$$R_2 = R_{total}^s + R_{total}^t \qquad (11)$$

**Step 11b: Estimate QHOs using data for selected year.** The QHOs that are to be used as the basis for comparison with results for safety goal QHO risk metrics are calculated using Equations (12) and (13):

### Early Fatality Risk QHO ($QHO_{EF}$)

$$QHO_{EF} = \left(\frac{1}{1000}\right) \cdot \left(\frac{accidental\ deaths}{US\ population}\right)^v \qquad (12)$$

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[u] Hereafter, the term "*safety goal QHO risk metrics*" is used to collectively refer to the total mean risk results obtained by combining mean frequency estimates with mean values for the two selected conditional consequence metrics calculated by MACCS: (1) average individual early fatality risk at one mile from the NPP site boundary; and (2) population-weighted latent cancer fatality risk within ten miles of the NPP site boundary.

[v] Based on 2013 data from the National Vital Statistics System[42] and the U.S. Census Bureau,[43] the sum of prompt fatality risks resulting from unintentional accidents for the U.S. population is 4.13E-04 per year—which results in an early fatality risk QHO of 4.13E-07 per year.

**Latent Cancer Fatality Risk QHO ($QHO_{LCF}$)**

$$QHO_{LCF} = \left(\frac{1}{1000}\right) \cdot \left(\frac{cancer\ deaths}{US\ population}\right)^{\text{w}} \qquad (13)$$

**Step 12: Estimate QHO margin for each QHO and each safety goal policy alternative.** The margin to each QHO is calculated for each policy alternative using Equations (14) and (15):

**Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included ($M_1$)**

$$M_1 = \frac{QHO}{R_1} \qquad (14)$$

**Option 2: Single-Unit and Two-Unit Accident Scenarios Included ($M_2$)**

$$M_2 = \frac{QHO}{R_2} \qquad (15)$$

**Step 13: Estimate figures of merit for comparison of safety goal policy alternatives for each QHO.** The figures of merit to be used for comparison of policy alternatives and evaluation of the effect of including the contribution from two-unit accident scenarios to safety goal QHO risk metrics are calculated for each policy alternative using Equations (16) and (17):

**Figure of Merit 1: Percent Change in Risk Metric ($FOM_1$)**

$$FOM_1 = \left(\frac{R_2 - R_1}{R_1}\right) \cdot 100\% \qquad (16)$$

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w Based on 2013 data from the National Vital Statistics System[42] and the U.S. Census Bureau, [43] the sum of cancer fatality risks resulting from all causes among the U.S. population is 1.85E-03 per year—which results in a latent cancer fatality risk QHO of 1.85E-06 per year.

**Figure of Merit 2: Percent Change in QHO Margin ($FOM_2$)**

$$FOM_2 = \left(\frac{M_2 - M_1}{M_1}\right) \cdot 100\% \qquad (17)$$

## III.C.   Generalizability of Approach and Methods

Although the approach and computational methods have been developed to enable estimation of the contribution to safety goal QHO risk metrics from single-unit and two-unit accident scenarios for representative two-unit NPP sites, this approach is generalizable and could be expanded in application to include:

- Additional risk metrics that characterize other types of public risks from accidents involving NPP sites (e.g., societal risks of radiological health effects, protective action impacts on the affected population, environmental impacts from land contamination, and economic costs); and

- Estimation of the contribution to risk metrics of interest from multi-unit accident scenarios that involve combinations of more than two-units at a shared site.

Expansion of the approach for application to NPP sites comprised of more than two units would require consideration of additional factors. Example factors include:

- **The conditional probabilities of accidents involving at least *n* units, given that an accident involving at least *n-1* units has occurred.** As shown in the two-unit case, these conditional probabilities that reflect the level of dependence between co-located units influence two estimated quantities: (1) the multi-unit accident scenario frequency used to calculate the unadjusted multi-unit accident scenario risk; and (2) the multi-unit frequency adjustment factor used to adjust multi-unit accident scenario risk estimates to account for the frequency

contribution from multi-unit accident scenarios that are not modeled, but that are assumed to be represented by the class of multi-unit accident scenarios that have been modeled. The Multiple Greek Letter (MGL) method for treatment of CCF events[32] could be adapted using operational experience data in the LER database to develop estimates for applicable conditional probabilities.

- **Potential inter-unit differences in design or differences in timing of accident initiation, progression, or radiological releases.** If important differences between units exist, the choice of which unit serves as the reference unit can impact results. In addition, results from the two-unit case will demonstrate that differences in the timing of accident initiation, progression, and radiological releases between co-located units can have different effects depending on which unit is selected as the reference unit. Under these circumstances, the order in which a multi-unit accident occurs can be important. Therefore, the number of ways in which a multi-unit accident involving $n$ units can occur would be more appropriately defined by the number of permutations (ordered combinations) rather than combinations.

The expansion and application of these methods to multi-unit sites comprised of more than two units requires consideration and treatment of additional factors that necessarily complicate the analysis. Application of these methods to NPP sites comprised of different types of units with fundamentally different radionuclide inventories and risk profiles would require consideration and treatment of even more factors that would complicate the analysis. However, it is envisioned that existing methods can be adapted to address these additional complexities, and do not foresee any insurmountable barriers to adapting this approach to more diverse NPP sites.

# Chapter IV: Two-Unit Case Study Results and Discussion

## IV.A.  Summary Results for Both Representative NPP Sites

Summary results that address all research aims—including the base case analysis

for Aim 1 and the one-way sensitivity analyses for Aim 2 and Aim 3—are presented

in this section for both representative NPP sites. Section IV.B. presents additional

results generated from the base case analysis at more detailed and site-specific

levels for each representative NPP site.

### IV.A.1. Base Case Analysis: Effect of Including Contribution from Two-Unit Accident Scenarios on Safety Goal QHO Risk Metrics and QHO Margins

Table V summarizes the results for each figure of merit by reactor-containment

design and safety goal QHO risk metric for the base case analysis. These results

show that including the contribution from two-unit accident scenarios results in:

1. **Non-negligible increases in risk.** Including the contribution from two-unit
   accidents resulted in 15% to 77% increases in the mean values of the safety goal
   QHO risk metrics for the representative BWR,  and 18% to 20% increases in the
   mean values of the safety goal QHO risk metrics for the representative PWR.

2. **Non-negligible reductions in QHO margin.** Including the contribution from two-
   unit accidents resulted in 13% to 43% reductions in mean margin to QHOs for
   the representative BWR, and 16% to 17% reductions in mean margin to QHOs
   for the representative PWR.

Table V also shows that figures of merit based on the average individual early fatality

risk metric appear to be more sensitive to the effect of including the contribution from

two-unit accident scenarios than those based on the population-weighted latent

cancer fatality risk metric. Moreover, the magnitude of this effect with respect to early

fatality risk appears to be greater for the representative BWR site (Peach Bottom)

than for the representative PWR site (Surry).

**Table V. Summary Results for Each Figure of Merit by Safety Goal QHO Risk Metric and Reactor-Containment Design for Base Case Analysis**

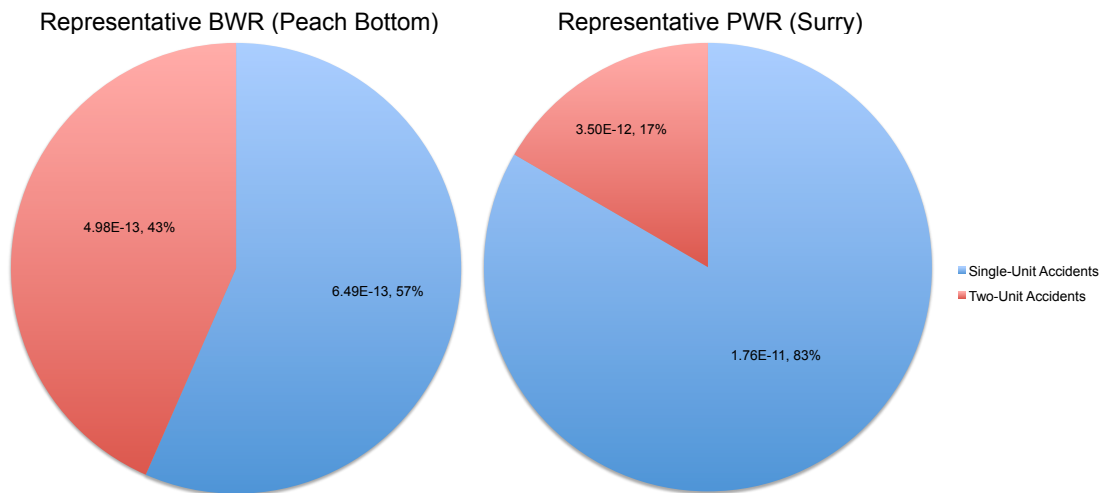| Safety Goal QHO Risk Metric | Figure of Merit 1 (% change risk)[a] | Figure of Merit 2 (% change QHO margin)[a] |
|---|---|---|
| *Representative BWR (Peach Bottom) Analysis* | | |
| Average Individual Early Fatality Risk (1 mi) | 77% | -43% |
| Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | 15% | -13% |
| *Representative PWR (Surry) Analysis* | | |
| Average Individual Early Fatality Risk (1 mi) | 20% | -17% |
| Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | 18% | -16% |
| [a] Each figure of merit represents a percent change that compares results with both single-unit and multi-unit accident scenario contributions included, relative to only single-unit accident contributions included. | | |



Representative BWR (Peach Bottom)    Representative PWR (Surry)

4.98E-13, 43%    6.49E-13, 57%    3.50E-12, 17%    1.76E-11, 83%

Single-Unit Accidents
Two-Unit Accidents

**Figure 10. Relative Contributions of Single-Unit Accidents and Two-Unit Accidents to Total Mean Average Individual Early Fatality Risk (1 mi).** Two-unit accidents contribute more to total early fatality risk relative to single-unit accidents for the representative BWR site than for the representative PWR site.
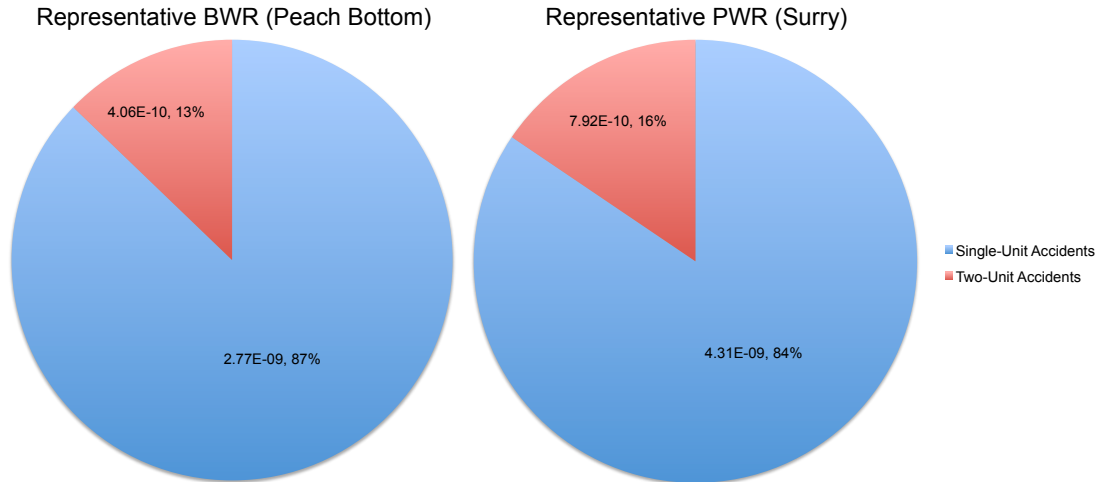
**Figure 11. Relative Contributions of Single-Unit Accidents and Two-Unit Accidents to Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi).** The relative contributions of single-unit accidents and two-unit accidents to total latent cancer fatality risk are approximately the same for the representative BWR site and the representative PWR site.

Figure 10 displays the relative contributions of single-unit accidents and two-unit accidents to total early fatality risk for both the representative BWR site and the representative PWR site. Figure 11 displays similar results with respect to total latent cancer fatality risk. Taken together, these figures suggest that early fatality risk is more sensitive to the effect of including the contribution from two-unit accidents than latent cancer fatality risk because two-unit accidents contribute a greater percentage to total early fatality risk relative to single-unit accidents for each NPP site than they contribute to total latent cancer fatality risk. Moreover, it appears that the magnitude of the effect of including the contribution from two-unit accident scenarios to early fatality risk is greater for the representative BWR site than for the representative PWR site because two-unit accidents comprise a greater percentage of total early fatality risk for the BWR site (43% for the BWR versus 17% for the PWR). By contrast, the two representative sites have comparable risk profiles with respect to latent cancer fatality risk, thereby resulting in similar effect estimates using figures of merit based on the population-weighted latent cancer fatality risk metric.

*IV.A.2. One-Way Sensitivity Analyses: Effect of Variation in Assumed Level of Inter-Unit Dependence*

Table VI summarizes results obtained for each figure of merit by reactor-containment design and safety goal QHO risk metric by performing one-way sensitivity analyses in which the assumed value of $\beta$—the global conditional probability of an accident occurring in the co-located unit, given that an accident occurs in the reference unit at a specified frequency—was varied over its full range of possible values from 0 to 1. Concurrent accidents involving the co-located and reference units were assumed to occur simultaneously as in the base case.

**Table VI. Summary Results for Effect of Variation in Assumed Level of Inter-Unit Dependence**

| Safety Goal QHO Risk Metric | Figure of Merit 1 (% change risk)[a] | | Figure of Merit 2 (% change QHO margin)[a] | |
|---|---|---|---|---|
| | $\beta = 0.1$[b] | $\beta = 1$ | $\beta = 0.1$[b] | $\beta = 1$ |
| *Representative BWR (Peach Bottom) Analysis* | | | | |
| Average Individual Early Fatality Risk (1 mi) | 77% | 770% | -43% | -89% |
| Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | 15% | 150% | -13% | -60% |
| *Representative PWR (Surry) Analysis* | | | | |
| Average Individual Early Fatality Risk (1 mi) | 20% | 200% | -17% | -67% |
| Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | 18% | 180% | -16% | -65% |
| [a] Each figure of merit represents a percent change that compares results with both single-unit and multi-unit accident scenario contributions included, relative to only single-unit accident contributions included. | | | | |
| [b] For $\beta = 0$, which is equivalent to assuming there are no inter-unit dependencies that could give rise to two-unit accidents given that single-unit accidents occur in the reference unit at a specified frequency, including the contribution from two-unit accidents to safety goal QHO risk metrics has no effect. These trivial results are not displayed here to conserve space. | | | | |

These results appear to reinforce the findings from the base case analysis. Increasing the value of $\beta$ effectively increases the frequency element of the risk contribution from two-unit accident scenarios, thereby increasing the relative importance of two-unit accidents compared to single-unit accidents. We would therefore expect the effect of including the contribution from two-unit accident scenarios to increase as $\beta$ increases, and this is what we observe.

Coupled with the results displayed in Figures 12 and 13—which respectively illustrate the effect of variation in β on the figures of merit for early fatality risk and latent cancer fatality risk for both representative NPP sites—we also observe that the figure of merit for percent change in risk appears to be more sensitive to the effect of variation in β than the figure of merit for percent change in QHO margin. Whereas the percent increase in risk increases linearly as β increases, the percent reduction in QHO margin increases less than linearly as β increases. As shown in Section IV.B., several orders of magnitude in margin to both QHOs exist for the base case analysis. Therefore, while increasing β to its maximum possible value of 1 causes an order of magnitude increase in the percent change in risk, the absolute change in risk is still small relative to the QHO margin. Thus a smaller effect is observed with respect to the percent change in QHO margin.
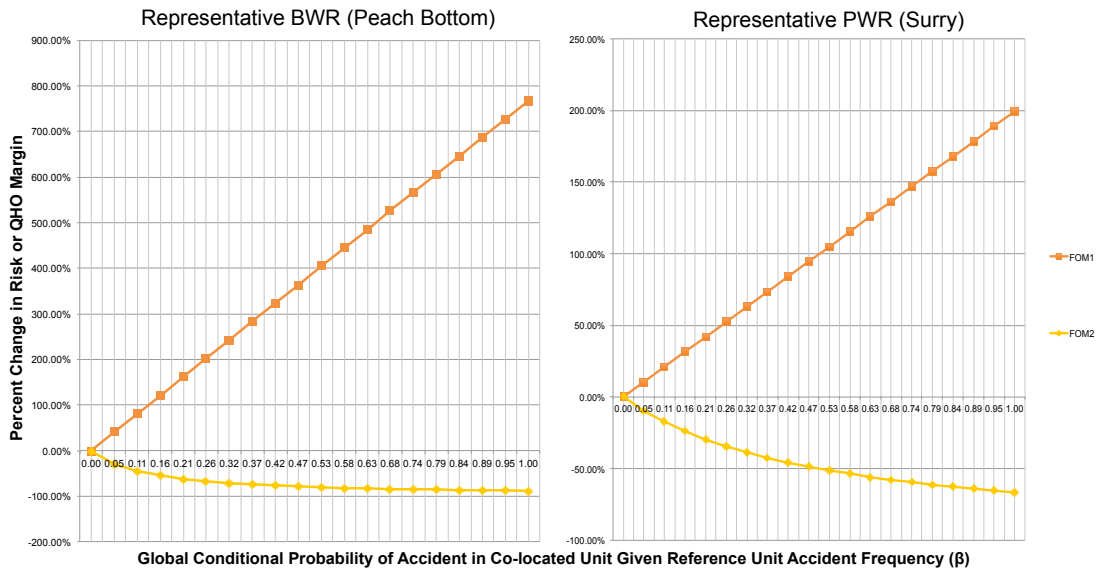


**Figure 12. Effect of Variation in Assumed Level of Inter-Unit Dependence on Total Mean Average Individual Early Fatality Risk (1 mi).** Variation in the value of *β* has a greater effect on the figure of merit for percent change in early fatality risk (displayed in orange) than the figure of merit for percent change in margin to the early fatality risk QHO (displayed in yellow) for both representative NPP sites.
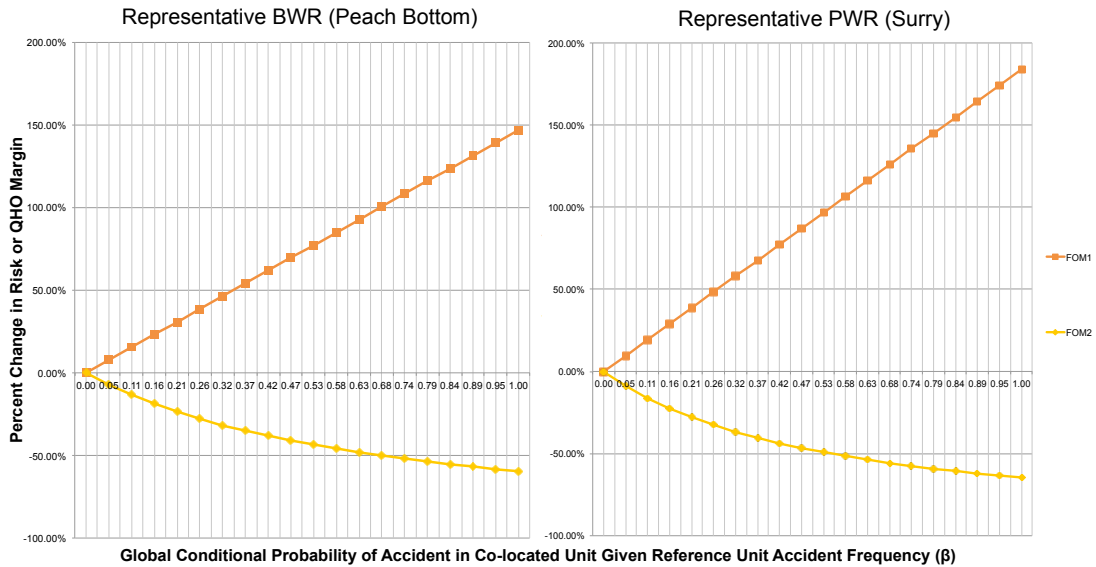
Representative BWR (Peach Bottom)          Representative PWR (Surry)

**Figure 13. Effect of Variation in Assumed Level of Inter-Unit Dependence on Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi).** Variation in the value of $\beta$ also has a greater effect on the figure of merit for percent change in latent cancer fatality risk (displayed in orange) than the figure of merit for percent change in margin to the latent cancer fatality risk QHO (displayed in yellow) for both representative NPP sites.

### IV.A.3. One-Way Sensitivity Analyses: Effect of Variation in Assumed Timing Offset Between Concurrent Two-Unit Accident Scenarios

Figures 14 through 17 illustrate the scenario-specific risk results from one-way sensitivity analyses that were performed to evaluate the effect of variation in the assumed timing offset between concurrent two-unit accident scenarios. As in the base case, the value of $\beta$ was assumed to be 0.10. Scenario-specific risk results are presented in lieu of results for the figures of merit because variation in the assumed timing offset resulted in a negligible effect on the figures of merit, while scenario-specific risk results highlight interesting findings from the one-way sensitivity analyses.

Figures 14 and 15 show that two-unit accident scenarios that include a more rapidly progressing STSBO scenario (STSBO-Base for the representative BWR analysis and STSBO-TISGTR for the representative PWR analysis) as a constituent accident

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scenario in the co-located unit exhibit a rapid decline in the average individual early fatality risk metric as the timing offset between concurrent accidents is varied from 0 (the assumed simultaneous accidents in both units for the base case) to a one-day offset between the co-located unit and reference unit accident scenarios. However, the same pattern is not observed for the mirror-image two-unit accident scenarios that have these more rapidly progressing STSBO scenarios as a constituent accident scenario in the reference unit. Whereas the mirror-image scenarios have identical results for the simultaneous case—which is expected based on the key assumption of identical units—their results diverge as the timing offset is increased from 0 days. In all of these cases, the average individual early fatality risk is significantly reduced as the timing offset is increased to effectively model alternative delay times for the more rapidly progressing scenario in the co-located unit, relative to the reference-unit accident scenario.

While not displayed, evaluation of the underlying data revealed that all of the early fatalities attributed to modeled two-unit accident scenarios occur among the 0.5% of the offsite population within ten miles of each NPP site that is assumed to not evacuate in response to each accident and instead maintains normal activity. As the timing offset is increased to model greater delay times for the more rapidly progressing scenario in the co-located, the number of early fatalities predicted to occur among this non-evacuating population decreases. A number of factors could be contributing to the observed behavior, including potential interactions between the delay time, accident progression timing, and: (1) statistical variability in weather conditions; and/or (2) implementation of protective actions (e.g., dose-dependent relocation—which is still applied to the non-evacuating population).
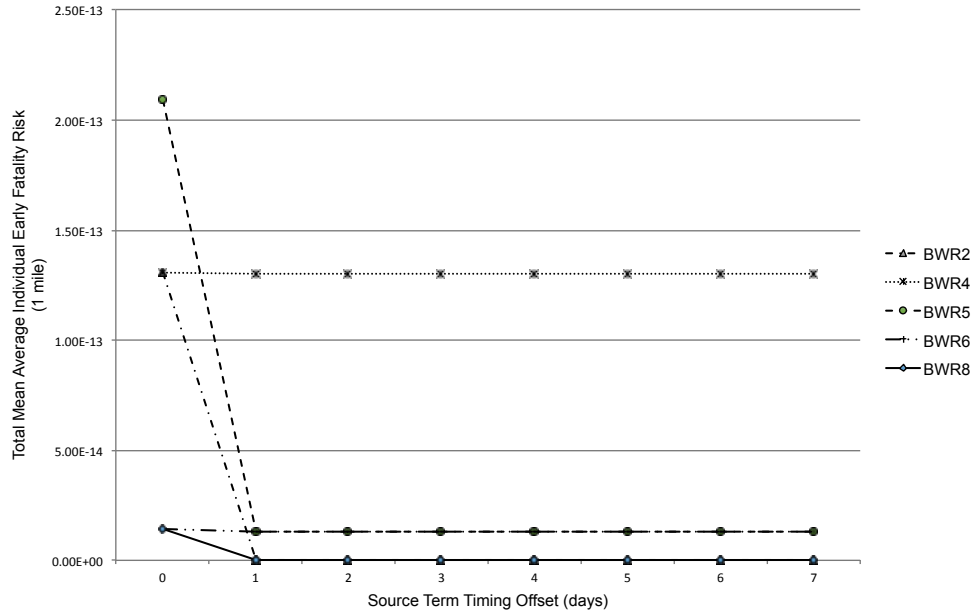
**Figure 14. Effect of Variation in the Assumed Timing Offset Between Concurrent Accidents on Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative BWR Site.** Increasing the delay time for more rapidly progressing accident scenarios in the co-located unit results in significant reductions in early fatality risk among the offsite population that is assumed to not evacuate for a representative BWR site.
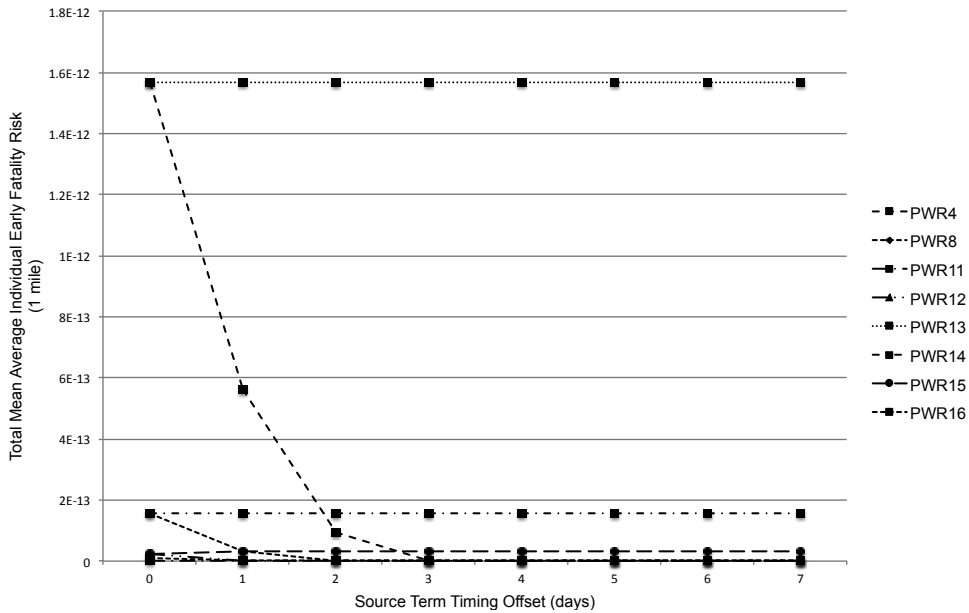


**Figure 15. Effect of Variation in the Assumed Timing Offset Between Concurrent Accidents on Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative PWR Site.** Increasing the delay time for more rapidly progressing accident scenarios in the co-located unit also results in significant reductions in early fatality risk among the offsite population that is assumed to not evacuate for a representative PWR site.

Figures 16 and 17 show that a different pattern is observed with respect to the total mean population-weighted latent cancer fatality risk metric. As shown in Figure 17, variation in the assumed timing offset between concurrent accidents in both units does not significantly impact latent cancer fatality risk for the representative PWR site. In general, this finding also applies to the representative BWR site. However, Figure 16 shows that the latent cancer fatality risk attributable to the class of two-unit accident scenarios represented by BWR1—which is comprised of LTSBO scenarios in both the co-located and reference units—actually increased as the timing offset was varied from 0 to 7 days.

While not displayed, evaluation of the underlying data revealed that all of the increase in the population-weighted latent cancer fatality risk for this scenario is attributed to increases in the risk attributed to long-term exposures that occur during the recovery phase of response to the two-unit accident. One possible explanation for this observation is that increasing the timing offset between constituent accident scenarios effectively provides more time for the weather conditions—especially wind direction—to change between releases of sequential plume segments. As a result, larger areas of land would be contaminated at lower levels of radioactivity. For this reason, larger areas of contaminated land would be able to be returned to habitability for a specified habitability criterion.[x] Thus, more individuals would be allowed to return home to land that has been contaminated, and would therefore be exposed to residual levels of contamination for the duration of the long-term phase. Since a dose-response model based on the linear no-threshold (LNT) hypothesis is used to calculate excess latent cancer fatalities attributable to the modeled accident scenarios, this increase in the number of individuals exposed to low levels of

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[x] The habitability criterion represents a maximum dose and an exposure period to receive that dose that is used to determine when contaminated land is deemed habitable.

radiation would necessarily result in a proportional increase in the population-weighted latent cancer fatality risk.[y] While this explanation seems reasonable, it cannot be used to explain why other two-unit accident scenarios that include the LTSBO scenario as a constituent do not exhibit similar behavior.

Taken together, these findings suggest that there may be synergistic effects or interactions between the assumed timing offset between concurrent accident scenarios, the timing of accident progression for the constituent accident scenarios, and other factors. However, additional research would be needed to formulate and test alternative hypotheses to obtain further insights into the underlying causes for this observed behavior.

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[y] In the SOARCA pilot study, sensitivity analyses were performed to evaluate the impact of using plausible alternative dose-response models. Whereas the base case analyses used an LNT-based dose-response model, sensitivity cases used alternative dose truncation models for which the excess lifetime cancer cases attributable to modeled accidents were not quantified below specified dose levels. For this thesis research, only the LNT-based dose-response model from the base case analyses was used; no additional sensitivity analyses were performed to evaluate the impact of using plausible alternative dose-response models.
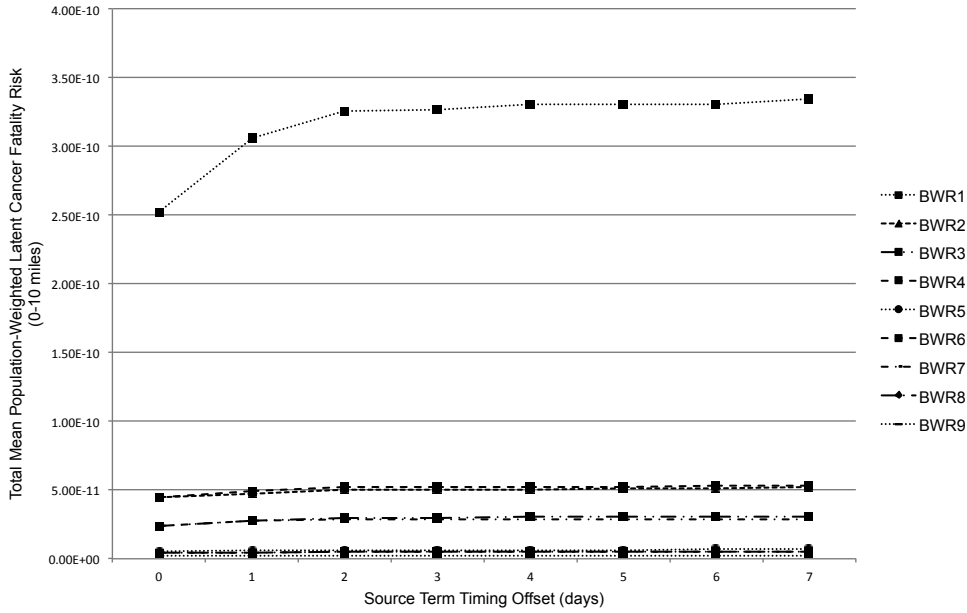
**Figure 16. Effect of Variation in the Assumed Timing Offset Between Concurrent Accidents on Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative BWR Site.** In general, variation in the timing offset between concurrent accident scenarios does not significantly impact latent cancer fatality risk for a representative BWR site. Where an increasing trend is observed, the increase is entirely attributed to latent cancers arising from long-term exposures during the recovery phase.
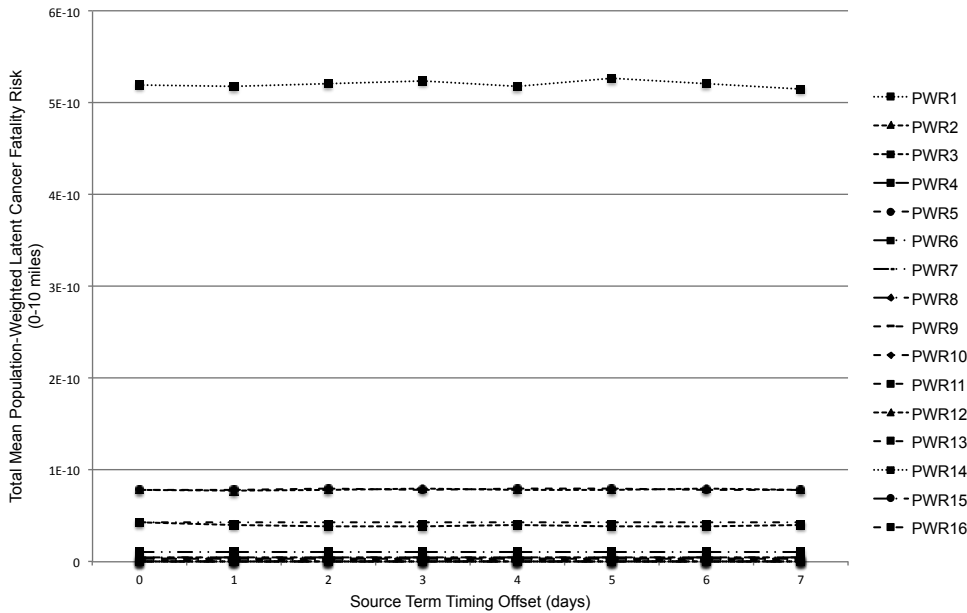


**Figure 17. Effect of Variation in the Assumed Timing Offset Between Concurrent Accidents on Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative PWR Site.** Variation in the timing offset between concurrent accident scenarios does not significantly impact latent cancer fatality risk for a representative PWR site.

## IV.B. More Detailed Site-Specific Results for the Base Case Analysis

Whereas Section IV.A. presents summary results that address all research aims for both representative NPP sites, this section presents additional results generated from the base case analysis at more detailed and site-specific levels for each representative NPP site and each safety goal QHO.

### IV.B.1. Representative BWR (Peach Bottom) Base Case Analysis

#### Early Fatality Risk QHO

Figures 18 and 19 respectively illustrate: (1) the distribution of adjusted single-unit risk contributions to the average individual early fatality risk metric for all single-unit accident scenarios modeled for Peach Bottom as part of the SOARCA pilot study; and (2) the distribution of adjusted two-unit risk contributions to the average individual early fatality risk metric for all two-unit accident scenarios modeled for this research. Figure 18 shows that the only class of single-unit accident scenarios to contribute to average individual early fatality risk at one-mile from the NPP site boundary is the class represented by the unmitigated and rapidly progressing STSBO-Base scenario. This finding is reasonable, considering the shorter period of time within this class of accident scenarios for Offsite Response Organizations (OROs) to implement protective actions that aim to reduce radiological dose and avoid deterministic health effects such as early injuries and fatalities arising from acute exposures to high levels of radiation.
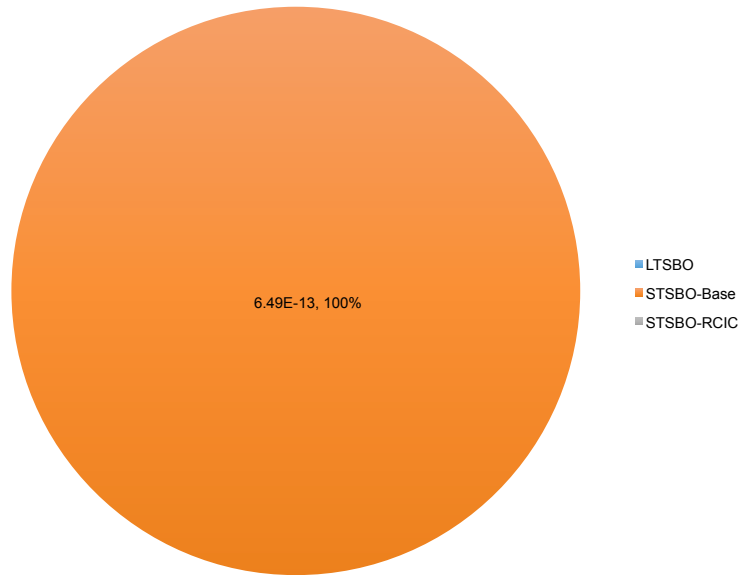
**Figure 18. Distribution of Adjusted Single-Unit Accident Scenario Risk Contributions to Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative BWR Site.** Single-unit accident scenarios represented by the unmitigated and rapidly progressing STSBO scenario are the only class of single-unit accident scenarios to contribute to early fatality risk for the representative BWR site.



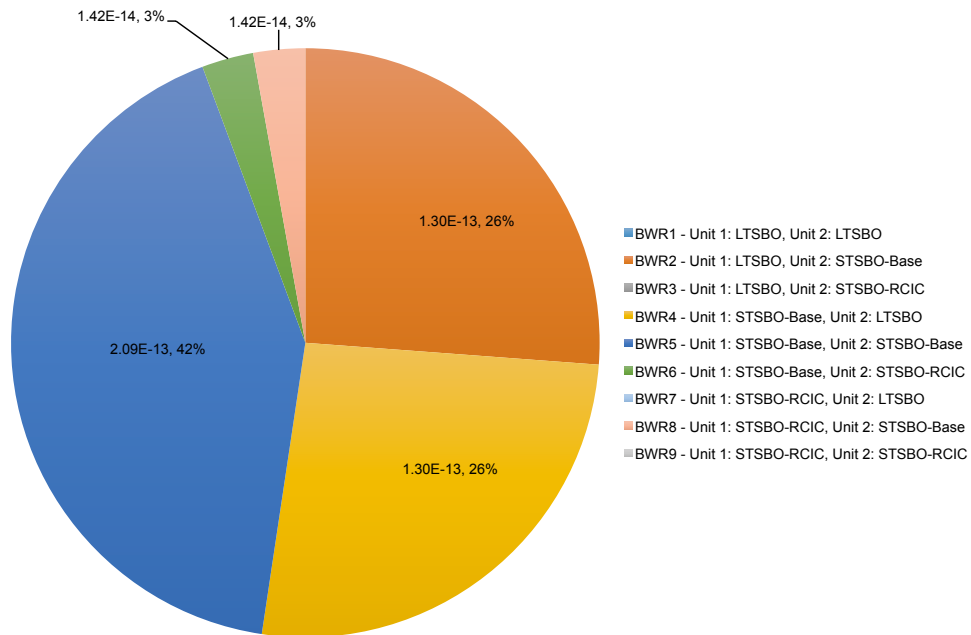**Figure 19. Distribution of Adjusted Two-Unit Accident Scenario Risk Contributions to Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative BWR Site.** Two-unit accident scenarios represented by scenarios that include the unmitigated and rapidly progressing STSBO scenario as a constituent are the dominant contributors to early fatality risk from two-unit accidents for the representative BWR site.

Figure 19 shows that the classes of two-unit accident scenarios that dominate the contributions to average individual early fatality risk at one-mile from the NPP site boundary are those classes that include the STSBO-Base scenario as a constituent accident scenario in either or both the reference unit and the co-located unit. Moreover, the class of two-unit accident scenarios represented by BWR 5—which is comprised of simultaneous STSBO-Base scenarios in both units—is the dominant contributor to two-unit accident risk with respect to average individual early fatality risk, comprising 42% of the total two-unit risk. This finding shows that accident progression and radiological release timing with respect to implementation of protective actions to avoid radiological dose is also important in the context of multi-unit accident scenarios.

Table VII summarizes the base case analysis results for the figures of merit that are used to evaluate the effect of expanding the scope of the safety goal policy to include the contribution to risk from multi-unit accident scenarios. As shown, combining the single-unit and two-unit contributions to average individual early fatality risk results in: (1) a 77% increase in risk; and (2) a 43% reduction in margin to the QHO.

**Table VII. Base Case Analysis Results with Respect to the Early Fatality Risk QHO for a Representative BWR Site.**

| Policy Alternative / Figure of Merit | Total Mean Average Individual Early Fatality Risk (1 mi) | QHO Margin |
|---|---|---|
| Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included | 6.E-13 | 6.E+05 |
| Option 2: Single-Unit and Two-Unit Accident Scenarios Included | 1.E-12 | 4.E+05 |
| Figures of Merit: Percent Change Option 2 Relative to Option 1 | 77% | -43% |

**Latent Cancer Fatality Risk QHO**

Figures 20 and 21 respectively illustrate: (1) the distribution of adjusted single-unit risk contributions to the population-weighted latent cancer fatality risk metric for all single-unit accident scenarios modeled for Peach Bottom as part of the SOARCA pilot study; and (2) the distribution of adjusted two-unit risk contributions to the population-weighted latent cancer fatality risk metric for all two-unit accident scenarios modeled for this research. Figure 20 shows that the dominant class of single-unit accident scenarios that contribute to population-weighted latent cancer fatality risk within 10 miles of the NPP site boundary is the class represented by the LTSBO scenario, which comprises 77% of the total single-unit accident risk for this metric. This finding is reasonable, considering two effects the longer release durations can have on offsite response to this class of scenarios:

1. The longer period of time for OROs to implement protective actions that reduce early phase radiological exposures—which increases the probability that individuals who receive early phase doses will die from cancer over the course of their lifetime, while making it less likely that they will die from acute exposures to high levels of radiation over short periods of time.

2. The longer release durations also provide more time for the weather conditions— especially wind direction—to change between releases of sequential plume segments. As a result, larger areas of land would be contaminated at lower levels of radioactivity. For this reason, larger areas of contaminated land would be able to be returned to habitability for a specified habitability criterion. Thus, more individuals would be allowed to return home to land that has been contaminated, and would therefore be exposed to residual levels of contamination for the

duration of the long-term phase. Since a dose-response model based on the LNT hypothesis is used to calculate excess latent cancer fatalities attributable to the modeled accident scenarios, this increase in the number of individuals exposed to low levels of radiation will result in a proportional increase in the population-weighted latent cancer fatality risk.



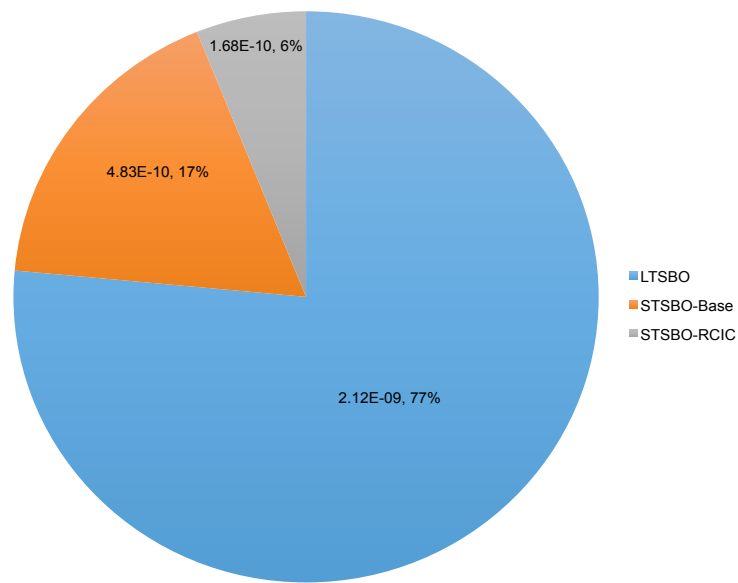**Figure 20. Distribution of Adjusted Single-Unit Accident Scenario Risk Contributions to Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative BWR Site.** Single-unit accident scenarios represented by the more slowly progressing LTSBO scenario are the dominant contributors to latent cancer fatality risk from single-unit accidents for the representative BWR site.
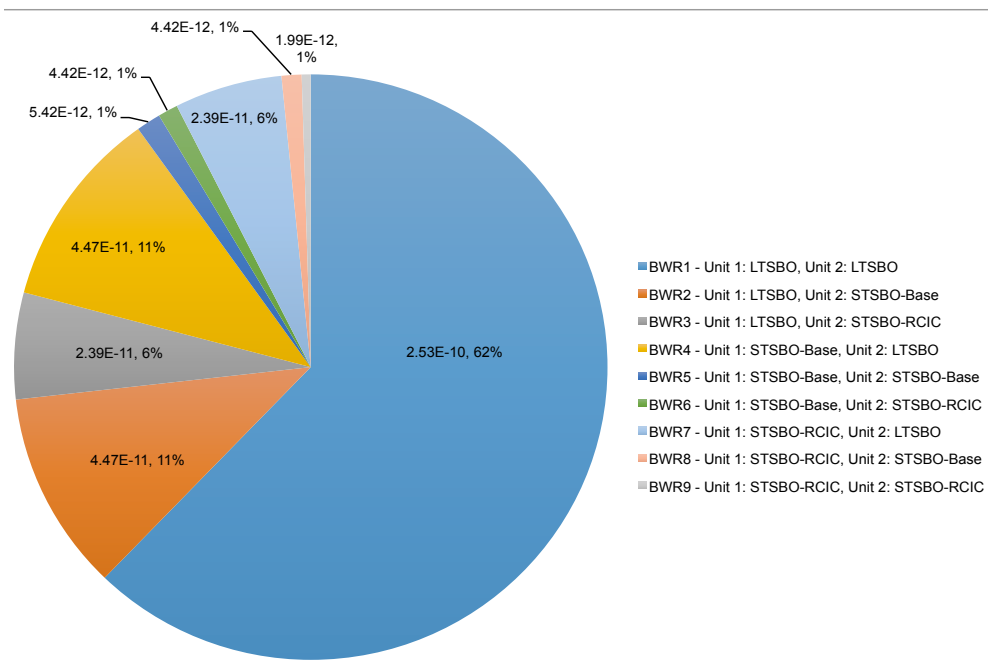
**Figure 21. Distribution of Adjusted Two-Unit Accident Scenario Risk Contributions to Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative BWR Site.** Two-unit accident scenarios represented by scenarios that include a more slowly progressing LTSBO scenario as a constituent are the dominant contributors to latent cancer fatality risk from two-unit accidents for a representative BWR site.

Figure 21 shows that the classes of two-unit accident scenarios that dominate the contributions to population weighted latent cancer fatality risk within 10 miles from the NPP site boundary are those classes that include the LTSBO scenario as a constituent accident scenario in either or both the reference unit and the co-located unit. Moreover, the class of two-unit accident scenarios represented by BWR 1— which is comprised of simultaneous LTSBO scenarios in both units—is the dominant contributor to two-unit accident risk with respect to population-weighted latent cancer fatality risk, comprising 62% of the total two-unit risk. This finding shows that accident progression and radiological release timing with respect to increasing dispersion of radionuclides and the size of the population exposed to low-levels of radioactive contamination for long periods of time upon returning home are also important factors in the context of multi-unit accident scenarios.

Table VIII summarizes the base case analysis results for the figures of merit that are used to evaluate the effect of expanding the scope of the safety goal policy to include the contribution to risk from multi-unit accident scenarios. As shown, combining the single-unit and two-unit contributions to population-weighted latent cancer fatality risk results in: (1) a 15% increase in risk; and (2) a 13% reduction in margin to the QHO.

**Table VIII. Base Case Analysis Results with Respect to the Latent Cancer Fatality Risk QHO for a Representative BWR Site.**

| Policy Alternative / Figure of Merit | Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | QHO Margin |
|---|---|---|
| Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included | 3.E-09 | 7.E+02 |
| Option 2: Single-Unit and Two-Unit Accident Scenarios Included | 3.E-09 | 6.E+02 |
| Figures of Merit: Percent Change Option 2 Relative to Option 1 | 15% | -13% |

Contrasting the base case analysis results with respect to both the early fatality risk QHO and the latent cancer fatality risk QHO, early fatality risk appears to be more sensitive to the effect of including the contribution from multi-unit accident scenarios than latent cancer fatality risk. This finding is reasonable and illustrates a principal concern with respect to multi-unit accident scenarios: that including the contribution from such accidents can be more important with respect to offsite radiological consequences such as early fatalities that are subject to non-linear behavior arising from threshold effects.

## IV.B.1. Representative PWR (Surry) Base Case Analysis

**Early Fatality Risk QHO**

Figures 22 and 23 respectively illustrate: (1) the distribution of adjusted single-unit risk contributions to the average individual early fatality risk metric for all single-unit accident scenarios modeled for Surry as part of the SOARCA pilot study; and (2) the distribution of adjusted two-unit risk contributions to the average individual early fatality risk metric for all two-unit accident scenarios modeled for this research. Figure 22 shows that the only class of single-unit accident scenarios to contribute to average individual early fatality risk at one-mile from the NPP site boundary is the class represented by the ISLOCA scenario. This finding is reasonable, considering the failure event that causes bypass of the containment structure, resulting in larger amounts of radiological materials being released over shorter periods of time, before OROs can effectively implement protective actions.
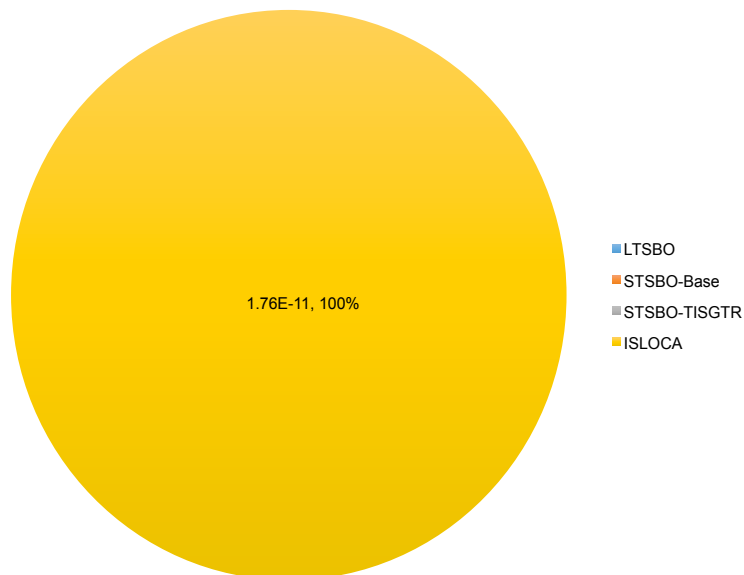


**Figure 22. Distribution of Adjusted Single-Unit Accident Scenario Risk Contributions to Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative PWR Site.** Single-unit accident scenarios represented by the ISLOCA scenario are the only class of single-unit accident scenarios to contribute to early fatality risk for a representative PWR site.
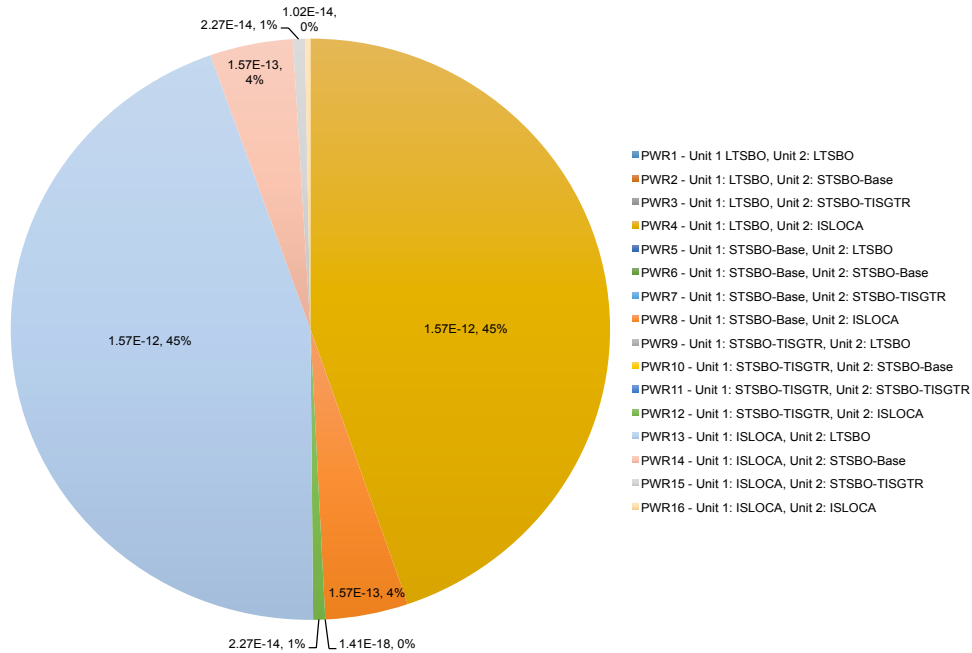
**Figure 23. Distribution of Adjusted Two-Unit Accident Scenario Risk Contributions to Total Mean Average Individual Early Fatality Risk (1 mi) for a Representative PWR Site.** Two-unit accident scenarios represented by scenarios that include a containment bypass scenario as a constituent are the dominant contributors to early fatality risk from two-unit accidents for a representative PWR site.

Figure 23 shows that the classes of two-unit accident scenarios that dominate the contributions to average individual early fatality risk at one-mile from the NPP site boundary are those classes that include a scenario involving containment bypass as a constituent accident scenario in either or both the reference unit and the co-located unit. These classes of scenarios comprise more than 98% of the total two-unit accident risk with respect to early fatality risk. This finding shows that containment failure or bypass events are also important in the context of multi-unit accident scenarios.

Table IX summarizes the base case analysis results for the figures of merit that are used to evaluate the effect of expanding the scope of the safety goal policy to include the contribution to risk from multi-unit accident scenarios. As shown, combining the single-unit and two-unit contributions to average individual early

fatality risk results in: (1) a 20% increase in risk; and (2) a 17% reduction in margin to the QHO. Contrasting these results with the corresponding results for the BWR analysis presented in Table VII, it is apparent that the average individual early fatality risk metric is more sensitive to the effects of including the contribution from multi-unit accident scenarios for the representative BWR site than for the representative PWR site. This difference is likely due to differences in the relative contributions of two-unit versus single-unit accidents to early fatality risk for the two NPP sites. As shown in Figure 10, two-unit accidents comprise a greater percentage of total early fatality risk for the representative BWR site (43%) than for the representative PWR site (17%).

**Table IX. Base Case Analysis Results with Respect to the Early Fatality Risk QHO for a Representative PWR Site.**

| Policy Alternative / Figure of Merit | Total Mean Average Individual Early Fatality Risk (1 mi) | QHO Margin |
|---|---|---|
| Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included | 2.E-11 | 2.E+04 |
| Option 2: Single-Unit and Two-Unit Accident Scenarios Included | 2.E-11 | 2.E+04 |
| Figures of Merit: Percent Change Option 2 Relative to Option 1 | 20% | -17% |

**Latent Cancer Fatality Risk QHO**

Figures 24 and 25 respectively illustrate: (1) the distribution of adjusted single-unit risk contributions to the population-weighted latent cancer fatality risk metric for all single-unit accident scenarios modeled for Surry as part of the SOARCA pilot study; and (2) the distribution of adjusted two-unit risk contributions to the population-weighted latent cancer fatality risk metric for all two-unit accident scenarios modeled for this research. Figure 24 shows that the dominant class of single-unit accident scenarios that contribute to population-weighted latent cancer fatality risk within 10 miles of the NPP site boundary is the class represented by the LTSBO scenario, which comprises 74% of the total single-unit accident risk for this metric. This finding is consistent with what was observed in the representative BWR analysis and relies on a similar interpretation of the results.



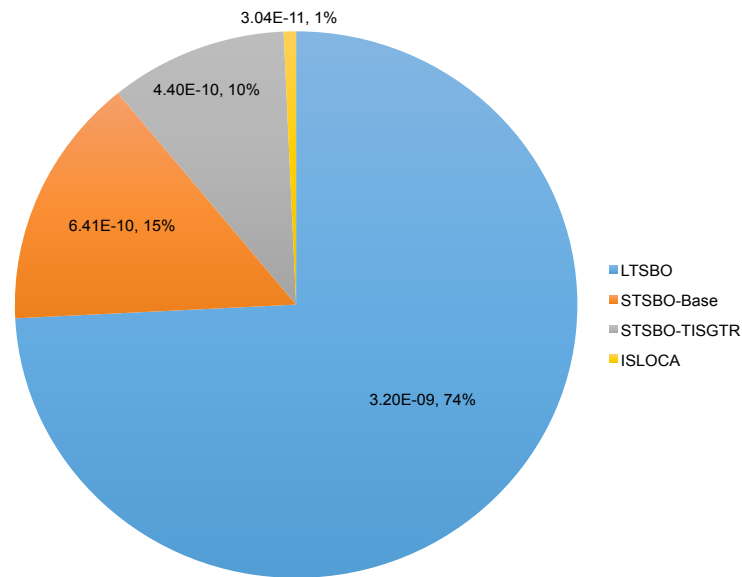**Figure 24. Distribution of Adjusted Single-Unit Accident Scenario Risk Contributions to Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative PWR Site.** Single-unit accident scenarios represented by the more slowly progressing LTSBO scenario are the dominant contributors to latent cancer fatality risk from single-unit accidents for a representative PWR site.
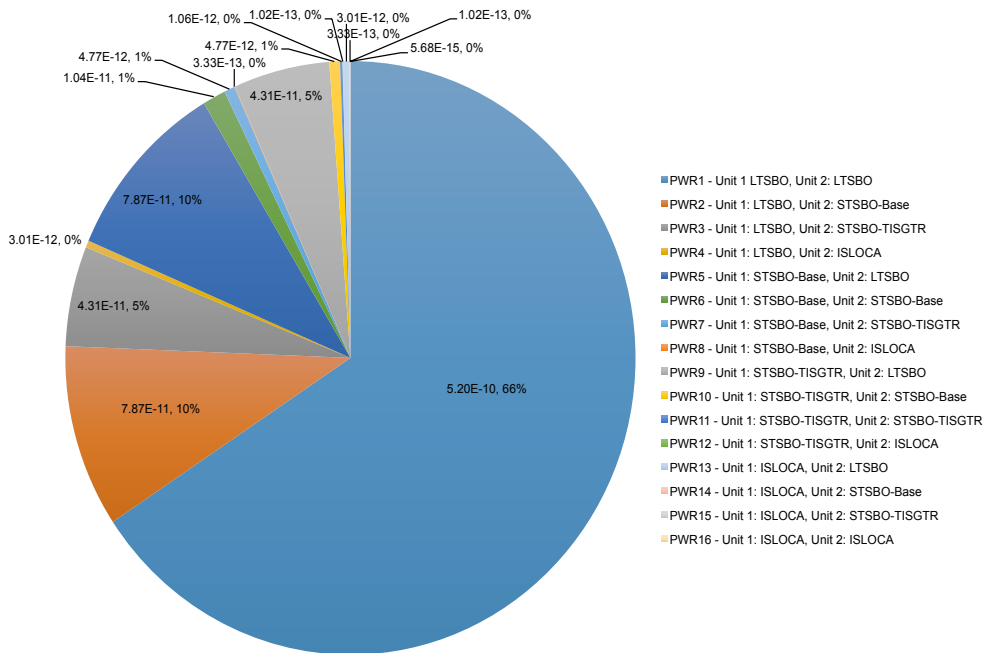
**Figure 25. Distribution of Adjusted Two-Unit Accident Scenario Risk Contributions to Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) for a Representative PWR Site.** Two-unit accident scenarios represented by scenarios that include a more slowly progressing LTSBO scenario as a constituent are the dominant contributors to latent cancer fatality risk from two-unit accidents for a representative PWR site.

Figure 25 shows that the classes of two-unit accident scenarios that dominate the contributions to population-weighted latent cancer fatality risk within 10 miles from the NPP site boundary are those classes that include the LTSBO scenario as a constituent accident scenario in either or both the reference unit and the co-located unit. Moreover, the class of two-unit accident scenarios represented by PWR 1—which is comprised of simultaneous LTSBO scenarios in both units—is the dominant contributor to two-unit accident risk with respect to population-weighted latent cancer fatality risk, comprising 66% of the total two-unit risk. This finding is consistent with what was observed in the representative BWR analysis and relies on a similar interpretation of the results.

Table X summarizes the base case analysis results for the figures of merit that are used to evaluate the effect of expanding the scope of the safety goal policy to

include the contribution to risk from multi-unit accident scenarios. As shown,

combining the single-unit and two-unit contributions to population-weighted latent

cancer fatality risk results in: (1) a 18% increase in risk; and (2) a 16% reduction in

margin to the QHO. Comparing these results with the corresponding results for the

BWR analysis presented in Table VIII, the notable difference between the

representative NPP sites with respect to the effect of including multi-unit accident

contributions to early fatality risk are not apparent in the effect estimates with respect

to latent cancer fatality risk. This is likely because the two NPP sites have similar

relative contributions of two-unit versus single-unit accidents to latent cancer fatality

risk. As shown in Figure 11, two-unit accidents comprise 13% of total latent cancer

fatality risk for the representative BWR site and 16% for the representative PWR site.

**Table X. Base Case Analysis Results with Respect to the Latent Cancer Fatality Risk QHO for a Representative PWR Site.**

| Policy Alternative | Total Mean Population-Weighted Latent Cancer Fatality Risk (0-10 mi) | QHO Margin |
|---|---|---|
| Option 1 (Status Quo): Only Single-Unit Accident Scenarios Included | 4.E-09 | 4.E+02 |
| Option 2: Single-Unit and Two-Unit Accident Scenarios Included | 5.E-09 | 4.E+02 |
| Figures of Merit: Percent Change Option 2 Relative to Option 1 | 18% | -16% |

Moreover, comparing the base case analysis results with respect to both the early

fatality risk QHO and the latent cancer fatality risk QHO shows that—in contrast to

what was observed for the representative BWR site—including the contribution from

multi-unit accident scenarios appears to have a similar effect on both early fatality

risk and latent cancer fatality risk for the representative PWR site. As shown in

Figures 10 and 11 respectively, this is likely because two-unit accidents contribute

similar percentages to total early fatality risk (17%) and total latent cancer fatality risk

(16%) for the representative PWR site.

# Chapter V:  Conclusions and Recommendations

## V.A.   Research Summary

The overall aim of this thesis research was to evaluate the effect of including the contribution from multi-unit accident scenarios to safety goal QHO risk metrics in safety goal policy applications. To do this, two policy alternatives were identified for evaluation: (1) a status quo option that reflects the existing application of the safety goal policy by including only the contribution from single-unit accident scenarios in estimating the safety goal QHO risk metrics; and (2) a second option that reflects a hypothetical expansion in the scope and application of the safety goal policy by including the contribution from multi-unit accident scenarios to these risk metrics. Two figures of merit were then developed to evaluate the effect of including the contribution from multi-unit accident scenarios to safety goal QHO risk metrics. These figures of merit accomplish this by quantifying the relative difference between the two policy alternatives with respect to: (1) the safety goal QHO risk metrics; and (2) the margin to the QHOs.

To leverage decades of severe accident research that has led to an improved understanding and modeling of severe accident and offsite radiological consequence phenomena, the SOARCA pilot study was selected as the foundation for this research to build upon. This extensively peer-reviewed PCA developed state-of-the-art models and integrated analyses of accident progression and offsite radiological consequences to estimate the realistic outcomes for a limited set of single-unit accident scenarios that were judged to be important for two NPP sites considered to be representative of a large group of NPP sites that use similar BWR and PWR reactor and containment designs.

Using the state-of-the-art consequence models for each of the single-unit accidents that were selected for detailed modeling and evaluation as part of the SOARCA pilot study, a set of 25 two-unit accident scenario consequence models were developed, with one unit serving as the reference unit with respect to the timing of accident initiation, progression, and radiological releases. For a base case analysis, each of these two-unit accident scenario models were run to estimate the conditional consequence contribution to safety goal QHO risk metrics assuming the reference unit and co-located unit accident scenarios occur simultaneously. For each modeled two-unit accident scenario, seven one-way sensitivity analyses were then performed to evaluate the effect of varying the assumed timing offset between concurrent two-unit accidents from 1 to 7 days in one-day increments; this resulted in a total of 200 two-unit accident simulations and sets of conditional consequence metrics for evaluation.

To evaluate the effects of including the contribution to safety goal risk metrics from two-unit accidents relative to including only the contribution from single-unit accidents, methods were developed for estimating the figures of merit using the conditional consequence results from state-of-the-art consequence models developed as part of the SOARCA pilot study and this research. A key assumption that underlies this approach is that the set of modeled single-unit and two-unit accident scenarios are representative of the broad spectrum of all possible single-unit and two-unit accident scenarios with respect to their conditional consequence contribution to the safety goal QHO risk metrics. This assumption allows for adjusting the scenario-specific risk results to develop an adjusted measure of risk that accounts for the contribution to the frequency element from accident scenarios in similar classes that have not been modeled.

In this approach, a global conditional probability of an accident in the co-located unit, given a specified reference unit accident frequency is used to estimate: (1) the frequency for each two-unit accident scenario; and (2) the frequency adjustment factor that is used to adjust risk measures to account for the unmodeled accident scenarios. A key assumption that underlies the use of this parameter is that a global average value can be applied across all two-unit accident scenarios. For the base case analysis, the value of this parameter was assumed to be 0.1. One-way sensitivity analyses were then performed to evaluate the effect of varying this parameter over its entire range of possible values from 0 to 1.

Although equations were developed and calculations were performed specifically for a two-unit case study involving two representative NPP sites, the overall approach and methods are generalizable and can be expanded to perform similar evaluations for NPP sites comprised of diverse and more than two co-located units.

## V.B.  Key Conclusions

Demonstration of the approach and methods to the two-unit case using two representative NPP sites generated a number of findings and insights that are documented in Chapter IV. Key conclusions developed from this research are:

1. **Including the contribution from multi-unit accidents results in non-negligible increases in safety goal QHO risk metrics and reductions in QHO margin.** The magnitude of the effect depends on the safety goal QHO risk metric under consideration and the reactor-containment design—with effect differences being driven by the site-specific relative contributions of single-unit accidents versus multi-unit accidents to the risk metric under consideration.

2. **The percent change in risk attributed to including the contribution from**

**multi-unit accidents is more sensitive to assumptions about the level of inter-unit dependence than the percent change in QHO margin.** This appears to be due to the several orders of magnitude in margin to both QHOs that exist even for the worst-case assumption of complete inter-unit dependence. Therefore, while including the contribution from multi-unit accidents to safety goal QHO metrics may result in non-negligible increases in risk estimates, this expansion in the scope of the safety goal policy may not result in different conclusions from safety goal evaluations.

3. **There may be synergistic effects between the assumed timing offset between concurrent accident scenarios and other factors that influence offsite radiological consequences (e.g., statistical variability in weather conditions and protective actions taken to reduce radiological dose).** Additional research is needed to formulate and test alternative hypotheses about the underlying causes of the observed behavior.

## V.C.    Research Limitations and Recommendations

insights derived from this research can be used to inform current USNRC and nuclear industry stakeholder deliberations about whether and to what extent the existing safety goal policy should be expanded to include consideration of the risk contribution from multi-unit accident scenarios. However, like any research study that aims to address questions pertaining to complex systems, limitations in study design and the need to make choices and assumptions with respect to model structure or parameter values can: (1) limit the extent to which findings can be generalized beyond the study population; and/or (2) introduce the potential for bias in results and their interpretation or application.

The overall design for this research study was summarized in Section I.E. Key assumptions that underlie this study design were then highlighted in Section I.F. Finally, Section I.G. describes the scope of the analysis, clearly identifying aspects that have been deliberately excluded from this analysis.

In this section, the limitations and assumptions that are judged to have the greatest potential for introducing bias or limiting the generalizability of study findings are highlighted, along with recommendations for future research to address known issues.

### V.C.1. Reactor-Containment Design and NPP Site Study Population

Since the SOARCA pilot study was selected as the foundation for this research to build upon, only 6% (2 out of 35) of the population of multi-unit NPP sites in the U.S. was selected for detailed modeling and evaluation as part of this research study. Yet these two NPP sites utilize reactor and containment designs similar to those used at sites that collectively represent 74% (26 out of 35) of U.S. multi-unit NPP sites. These two NPP sites have therefore been selected for inclusion in a number of previous NPP Level 3 PRA studies and PCA studies.

However, this also means that these two "representative" BWR and PWR sites may not adequately represent 25% (9 out of 35) of U.S. multi-unit NPP sites. Moreover, even for those NPP sites that share a similar reactor and containment design, there will always be site-specific or even unit-specific differences that can lead to differences in risk contributors. For these reasons, caution should be used in attempting to generalize the findings of this study to other NPP sites.

Additional research is therefore needed to apply the methods developed as part of this research to additional reactor and containment designs and/or NPP sites. Given

its continued use of best practices developed as part of the SOARCA pilot study, the follow-on SOARCA study for the Sequoyah Nuclear Plant[37] described in Section II.D.7 could serve as the foundation for a logical next step.

### V.C.2. Set of Modeled Accident Scenarios and Method for Estimating Risk Metrics

A collective total of 32 single-unit and two-unit accident scenarios (12 for a representative BWR and 20 for a representative PWR) were selected for detailed modeling and evaluation as part of this research study. The safety goal QHOs were developed for comparison with risk results for corresponding metrics from full-scope NPP Level 3 PRAs that model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential accident scenarios. To evaluate the effect of expanding the scope of the safety goal policy to include consideration of the risk contribution from multi-unit accident scenarios, methods were developed for estimating risk-based figures of merit using the conditional consequence results from the state-of-the-art consequence models developed as part of the SOARCA pilot study and this research. A key assumption underlying this approach is that the set of modeled single-unit and two-unit accident scenarios are representative of the broad spectrum of all possible single-unit and two-unit accident scenarios with respect to their conditional consequence contribution to the safety goal QHO risk metrics. This assumption allowed for adjusting the scenario-specific risk results to develop an adjusted measure of risk that accounts for the contribution to the frequency element from accident scenarios in similar classes that have not been modeled.

Additional research is needed to: (1) benchmark this method of using a frequency adjustment factor to adjust for the contribution to risk from accident scenarios that are not modeled; and (2) evaluate the key underlying assumption about the extent to

which a limited set of modeled accident scenarios can represent the full spectrum of potential accident consequences. Previously completed NPP Level 3 PRA studies (e.g. the NUREG-1150 study)[41] could provide a useful starting point, but are becoming increasingly dated—especially with respect to the offsite consequence models. Future NPP Level 3 PRA studies that leverage our improved understanding of severe accident progression and offsite radiological consequences would likely prove to be more useful. A potential candidate once it is completed and the results are made publicly available includes the USNRC's ongoing project to develop a full-scope, integrated site Level 3 PRA model for a U.S. multi-unit NPP site.[44] In addition to serving as a more contemporary reference, this study could potentially provide useful information about the applicability of this approach to NPP sites comprised of multiple and diverse radiological sources, including spent fuel pool units and dry cask storage facilities.

### V.C.3. Research Questions and Selected Risk Metrics

This research study was designed to evaluate the implications of expanding the safety goal policy to include consideration of the contribution from multi-unit accident scenarios to the safety goal QHO metrics. The figures of merit selected for evaluation were thus based solely on the safety goal QHO metrics.

There are many other research questions that could be studied with respect to the safety goal policy and the implications of expanding its scope. Of particular interest in the wake of the 2011 Fukushima nuclear accident are the potential implications of expanding the scope of the safety goal policy and QHOs to include additional risk metrics that can provide a more complete characterization of the public risks attributable to nuclear accidents than the safety goal QHO metrics alone can provide. Examples of such risk metrics include those that characterize: (1) societal risk of

radiological health effects (e.g., population dose and total numbers of radiological health effects); (2) protective action impacts on the affected population (e.g., total numbers of people impacted by long-term relocation); (3) environmental impacts (e.g., total areas of land contaminated at various levels of radioactivity); and (4) total economic costs.[45,46]

Additional research studies that aim to evaluate the effect of including contributions from multi-unit accident scenarios to this broader set of risk metrics could provide a more comprehensive characterization of: (1) the relative risks attributable to both single-unit and multi-unit accident scenarios; and (2) the potential health and safety benefits of proposed regulatory actions that may reduce the level of residual risk to the public from nuclear accidents in ways the existing set of safety goal QHO metrics are unable to measure.

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