RISK ANALYSIS OF MULTI-UNIT ACCIDENT EFFECTS ON PUBLIC HEALTH AND SAFETY USING STATE-OF-THE-ART MODELS: U.S. NUCLEAR REGULATORY COMMISSION SAFETY GOAL POLICY IMPLICATIONS

by

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A dissertation submitted to Johns Hopkins University in conformity with the requirements for the degree of Doctor of Philosophy

Baltimore, Maryland

May 2017

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This dissertation was prepared by an employee of the U.S. Nuclear Regulatory Commission (USNRC) on his own time apart from his regular duties. The USNRC has neither approved nor disapproved its technical content.
Abstract

The U.S. Nuclear Regulatory Commission (USNRC) safety goal policy defines an acceptable level of radiological risk to the public from nuclear power plant (NPP) operations. It addresses “How safe is safe enough?” by specifying qualitative safety goals and quantitative health objectives (QHOs) to measure safety goal attainment. Agency screening evaluations compare results from full-scope NPP probabilistic risk analyses (PRAs) with corresponding QHOs to determine whether proposed regulatory actions to further enhance NPP safety should be rejected before performing cost-benefit analyses to estimate their net benefit. Recent experience—including the 2011 Fukushima nuclear accident—highlights two gaps in the safety goal policy that can lead to underestimating or incompletely characterizing public risk and thus premature rejection of proposed safety enhancements. First, experience indicates concurrent accident scenarios involving multiple units at a shared site can occur with non-negligible frequency. Yet, the USNRC applies the safety goal policy to individual reactor units only and thus excludes the risk contribution from multi-unit accident scenarios for the nearly 75% of U.S. reactors located at multi-unit sites. Second, experience indicates radiological health risks to individuals living near NPP sites are negligible and indirect effects of protective actions that avert radiological dose are the dominant societal consequences of potential nuclear accident scenarios. Yet, QHOs used to measure safety goal attainment are limited to measures of individual risk of experiencing radiological health effects. This dissertation research evaluates the effects of expanding the safety goal policy to include: (1) multi-unit accident scenarios for multi-unit NPP sites; and (2) a broader set of public health risk metrics that includes measures of...
societal risk. To leverage decades of research and technology advances that have improved understanding and modeling of nuclear accident scenarios, state-of-the-art models developed for a contemporary peer-reviewed study are used to construct probabilistic models for a limited set of single-unit and multi-unit accident scenarios involving two representative NPP sites that each include two co-located reactor units. Efficient risk estimation models are then used to: (1) calibrate results from these models to account for the frequency contribution to risk from excluded accident scenarios; and (2) estimate approximate risk results, without having to develop a resource-intensive, contemporary full-scope NPP Level 3 PRA. Key conclusions from this dissertation research are: (1) including multi-unit accident scenarios increases risk by a non-negligible amount for all selected risk metrics, with multi-unit accident scenarios dominating risk at higher assumed levels of dependence between co-located units; (2) if only existing QHOs based on individual risk of experiencing radiological health effects continue to be used, including multi-unit accident scenarios would likely not impact results of USNRC screening evaluations for proposed regulatory actions, even under the worst-case assumption of complete inter-unit dependence; (3) relying solely on risk insights for single-unit accident scenarios can lead to flawed risk management strategies for risk metrics that involve threshold effects; (4) assuming multi-unit accident scenarios occur simultaneously may not be conservative for all risk metrics of interest; and (5) using a broader set of public health risk metrics provides a more complete characterization of public risks from potential nuclear accident scenarios, but further stakeholder engagement is needed to gauge interest in developing a broader set of QHOs and to judge the acceptability of additional risk results obtained through this research.
Acknowledgements

I will begin by acknowledging the important financial support I received from the U.S. Nuclear Regulatory Commission (USNRC) as part of its Graduate Fellowship Program. Under this program, I received three years of funding that covered all my education-related expenses, along with my full salary and benefits, while completing graduate courses and research as part of my Ph.D. program in the Johns Hopkins Bloomberg School of Public Health, Department of Health Policy and Management. During those formative years, I completed in-depth study of topics in risk sciences and public policy that enhanced my knowledge and understanding of these critical disciplines that provide the foundation for my dissertation research. I extend my genuine appreciation to the agency for its support during those three years that have greatly influenced my professional development and career path.

I also thank my supervisor, Dr. Kevin Coyne, Ph.D., P.E., for his confidence in selecting me for the probabilistic risk analysis (PRA) fellowship position and for his being open to my pursuing a graduate education in risk sciences and public policy from a school of public health—which is rather unusual, considering most agency PRA experts obtain a graduate degree in engineering, mathematics, or the physical sciences; I sincerely appreciate the flexibility he gave me in charting out my own course and in exploring my own research interests. I also appreciate the technical support and encouragement I received from Pat Santiago and her staff in the USNRC Office of Nuclear Regulatory Research, Division of Systems Analysis, Accident Analysis Branch who maintain expert knowledge of the state-of-the-art reactor consequence models that served as the foundation for this dissertation research to build upon.
I will be forever grateful for having the opportunity to work closely with my original advisor, Professor Tom Burke, Ph.D., M.P.H. As the Director of the Johns Hopkins Risk Sciences and Public Policy Institute, Tom’s reputation as an exceptional teacher and as a leader in advancing risk analysis and its application to complex environmental health policy problems, are what brought me to Hopkins. During my three years on campus, he served as my sounding board for ideas and struck an amazing balance between providing guidance and direction, while also giving me room to investigate my own interests. I will always value his mentorship and look forward to continuing our work together as friends and colleagues.

I will be forever indebted to my final co-advisors: Dr. Mary Fox, Ph.D., M.P.H. and Dr. Lainie Rutkow, Ph.D., J.D., M.P.H. Mary and Lainie graciously assumed their roles as my co-advisors when Tom began a leave of absence to pursue an amazing opportunity as the Science Advisor and Deputy Assistant Administrator for the Office of Research and Development at the U.S. Environmental Protection Agency. Mary brought important probabilistic analysis and environmental health risk assessment expertise to this work, while Lainie brought equally important public health law, policy, and emergency preparedness expertise. Both Mary and Lainie consistently provided prompt and constructive feedback that substantially improved the quality and potential impact of this dissertation research. I will always be grateful for their guidance and support that were instrumental in helping me see this work through to completion. I could not have done this without them.

I also extend my sincere thanks to the other members of my Thesis Advisory Committee: Dr. Dan Barnett, M.D., M.P.H. and Dr. Sauleh Siddiqui, Ph.D., who took time out of their busy schedules to review and constructively critique my
dissertation research. Dan brought valuable radiological and all-hazards public health emergency preparedness expertise to my committee, while Sauleh brought equally valuable applied mathematics and systems science expertise. Their insights and recommendations led to significant improvements to this work. I also thank Dr. Scott Levin, Ph.D. who—along with Tom Burke—graciously agreed to serve as an alternate member of my Final Oral Exam Committee. Special thanks go to Professor Peter Lees, Ph.D., who kindly agreed to review my dissertation and assumed the role of Committee Chair when Dan Barnett was unable to participate due to an extenuating family circumstance; his support on relatively short notice ensured I could defend my research as scheduled.

Finally, I would be remiss if I did not acknowledge the unconditional love and support I have received daily from my wife (Becca), my ten-year-old son (Jake), and my three-month-old son (Theo). While they often served as the most important and welcome distraction from making progress on this work, they also served as the primary motivation for my wanting to better myself and to make important contributions through my research. They continue to inspire me every day to be the best version of myself that I can be. I share this significant accomplishment with them. I am forever grateful for them, and for the furry members of our family (Murphy, Miles, and Ellie) who often required love and affection—or made messes around the house or yard—that further distracted me from making progress on this work.
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Chapter 1. Introduction

1.A. Safety Goals for Commercial Nuclear Power Plant Operations

The U.S. Nuclear Regulatory Commission (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 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 (hereafter Three Mile Island) 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 they achieved relative to the level of residual risk to the public from nuclear accidents.

The safety goal policy represents one aspect of the USNRC’s risk management philosophy; it addresses the question of “How safe is safe enough?” for regulatory decisions regarding NPP safety. This policy is a product of an agency effort to implement control mechanisms designed to avoid imposing costly regulatory requirements that may not be warranted relative to their safety benefit and the level

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* In this context, the term *residual risk* refers to the remaining risk that the public is exposed to when the USNRC has determined that an NPP has achieved a state of *adequate protection*. As described in Chapter 2, achieving a state of adequate protection is not equivalent to achieving a state of zero risk. While no explicit definition of what constitutes adequate protection exists, an NPP that complies with the set of relevant USNRC regulatory requirements is presumed to be in an adequate protection state. However, if an NPP is unable to comply with one or more USNRC regulatory requirements, this does not necessarily mean that the public is not adequately protected; the USNRC has sufficient flexibility to grant exemptions to its regulatory requirements on a case-specific basis, provided it determines that an adequate protection state can still be achieved with such exemptions in place.
of residual risk the public is exposed to from NPP operations. In practice, it guides agency screening evaluations to determine whether proposed regulatory actions that would impose additional generic requirements to enhance NPP safety beyond those needed to ensure adequate protection could provide a substantial safety benefit—relative to the level of residual risk to the public—to warrant further evaluation.

Using this approach, proposed regulatory actions that aim to further enhance NPP safety can be rejected before performing detailed cost-benefit analyses to determine whether they could be justified on their net present value (NPV) basis. In principle, rejection would occur if: (1) the potential safety benefit is judged to be not substantial enough; or (2) the residual risk to the public is determined to be at an acceptably low level, and thus limited resources that would be applied to the proposed regulatory action could be better applied to alternative courses of action.

The safety goal policy is based on a hierarchical framework comprised of two high-level qualitative safety goals that are each supported by a lower-level quantitative health objective (QHO). These QHOs 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 fatality risk (hereafter 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 latent cancer fatality risk).

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b By policy, safety goal screening evaluations are performed for proposed regulatory requirements that would be imposed generically on an affected class of NPPs, rather than on specific NPPs.

c Early fatality risks were previously termed prompt fatality risks. The latter terminology was still in use when the USNRC safety goal policy was developed and appears in Table I. However, the term early fatality risk is used hereafter to be consistent with its present use.
fatality risk QHO). The qualitative safety goals and their supporting QHOs are summarized in Table I.

### Table I. USNRC Qualitative Safety Goals and Quantitative Health Objectives

<table>
<thead>
<tr>
<th>Hierarchy Level</th>
<th>Type of Risk Addressed</th>
<th>Individual Risk</th>
<th>Societal Risk</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Level I:</strong> Qualitative Safety Goal</td>
<td></td>
<td>“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.”</td>
<td>“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.”</td>
</tr>
<tr>
<td><strong>Level II:</strong> Quantitative Health Objective</td>
<td>“The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.”</td>
<td>“The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.”</td>
<td></td>
</tr>
</tbody>
</table>

*For this purpose, the USNRC safety goal policy statement defines “in the vicinity of a nuclear power plant” as the area within one mile of an NPP site boundary.

*For this purpose, the USNRC safety goal policy statement defines “in the area near a nuclear power plant” as the area within ten miles of an NPP site boundary.

1.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 corresponding 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 likelihood of one or more adverse consequences 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?". In this framework, the risk attributable to accidents caused by potential failures involving complex engineered systems is characterized by a 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 assurance that important risk contributors are not missed.

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 assumed 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\textsuperscript{d}, 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\textsuperscript{e} conditional PCA results for each release category.\textsuperscript{4} For each consequence metric of interest, these frequency-weighted mean consequences are then summed across all radiological release categories to estimate the mean annual frequency of that consequence. Results from NPP Level 3 PRAs and from PCAs performed for other purposes can include mean values for: (1) average individual early fatality risk within one mile of the NPP site boundary; and (2) average individual latent cancer fatality risk within ten miles of the NPP site boundary.\textsuperscript{f} 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.\textsuperscript{g}

\textsuperscript{d} The different levels of analysis for traditional NPP PRAs are described in Chapter 2.
\textsuperscript{e} Common PCA tools for NPP applications can use probabilistic sampling techniques to account for what has been shown in previous studies to be dominant contributors to uncertainty in the offsite public health consequences for a given accidental release: (1) when the accident will occur within a specified time-period (typically one year of reactor operation for NPP applications); and (2) 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 over time. The mean conditional PCA results then represent the probability-weighted average for the selected consequence metric over all modeled weather conditions that are assumed to be possible when an accident occurs.
\textsuperscript{f} The USNRC safety goal policy specifies that mean values for QHO risk metrics are to be used for making safety-related decisions for safety goal policy applications.\textsuperscript{1}
\textsuperscript{g} Since the outputs of PCAs represent conditional results for these average individual risk measures, conditioned on the assumed occurrence of specified radiological releases, it is not appropriate to use such results to measure attainment of the qualitative safety goals. However, these conditional results can be and have been used to provide an additional perspective for interpreting PCA results within a safety goal QHO context.
1.C. Research Motivation

1.C.1. The Need to Account for Risk Contributions from Multi-Unit Accident Scenarios in Safety Goal Policy Applications

In the U.S., most 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. Put another way, nearly 75% (73 out of 99) of all operating reactors in the U.S. are located at multi-unit NPP sites that include two or three co-located reactor units. However, during development and evaluation of the safety goal policy, the USNRC decided that the safety goals and corresponding QHOs would be applied strictly on a per-reactor-unit basis, even for this majority of operating reactors located at 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. As a result, concurrent accident scenarios involving multiple co-located operating reactor units (hereafter multi-unit accident scenarios) have—with few exceptions—traditionally been excluded from NPP PRAs and safety goal screening evaluations in support of analyses of proposed regulatory actions.

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h Application of the safety goal policy on a per-reactor-unit basis means the policy is applied to individual reactor units at an NPP site, rather than collectively applying the policy to all reactor units at an NPP site. Moreover, single-unit NPP PRAs that provide input to safety goal screening evaluations for individual reactor units are performed under the key assumption that any reactor units co-located with the reactor unit of interest are in a safe and stable condition, thereby excluding multi-unit accident scenarios from the scope of the analysis.
Yet there are at least three compelling reasons for expanding the scope of NPP PRAs and safety goal screening 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)\(^7\) suggest that the contribution to reactor 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.\(^i\)

2. **Operational experience:** Lessons learned from the March 2011 accident at the Fukushima Daiichi Nuclear Power Station (FDNPS) in Japan\(^8\text{–}^{11}\)—together with findings from a recent review of an operational experience database for U.S. NPPs\(^12\)—demonstrate that adverse events (e.g., abnormal occurrences, incidents, or accidents) involving multiple operating reactor units co-located at a shared site occur at a non-negligible frequency.\(^j\)

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\(^i\) In probability theory, two events A and B are considered 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: 
\[
P(A|B) \neq P(A) \quad \text{and} \quad P(B|A) \neq P(B).
\]
Where dependencies exist, these can reduce or eliminate redundancy in design. Thus, 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.

\(^j\) Based on a recent review of U.S. operational experience data, Schroer developed a dependent failure event classification scheme to characterize potential dependencies across multiple units, so that single-unit PRA models could be integrated into a multi-unit PRA model for a shared NPP site that includes multiple co-located operating reactor units.
3. **Logical and ethical 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 corresponding QHOs.\(^1\) For multi-unit NPP sites, this population is not only exposed to the health and safety 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, which operational experience has shown occur with non-negligible frequency. It therefore stands to reason that the contributions to public risk from 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.](image)

Nearly 75% (73 out of 99) of operating reactors in the U.S. are located at multi-unit sites that include two or three reactor units.
This existing gap in the scope and application of the safety goal policy can have important implications. Since the risk contribution from multi-unit accident scenarios is 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 screening 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 inappropriately conclude that these proposed actions would not be justified based on a determination that: (1) the potential safety benefit is not substantial enough, or (2) the residual risk to the public is at an acceptably low level. These proposed actions would therefore be rejected before detailed cost-benefit analyses would be performed to determine whether the actions could result in a net benefit to society and thus improve societal welfare. In principle, including the contribution to residual risk from multi-unit accidents could result in retaining some proposed regulatory actions for detailed cost-benefit analyses that would otherwise be screened from further consideration if only the risk contribution from single-unit accident scenarios is included.

1.C.2. The Need for Including a Broader Set of Public Health Risk Metrics in Safety Goal Policy Applications

As shown in Table I, the QHOs used to measure attainment of the qualitative safety goals are limited to only two objectives. These objectives are for risks to average

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k This could arise especially for proposed regulatory actions that primarily aim to reduce the risk contribution from multi-unit accident scenarios (e.g., by addressing inter-unit dependencies that reduce or eliminate redundancy in design).
individuals in the vicinity of (or area near) NPPs of dying from acute tissue damage (or cancer) caused by radiation exposures resulting from accidental releases from these NPPs. Four issues arise from this limitation in the scope of the QHOs:

1. **The QHOs do not address what we now know to be the dominant public health risks attributable to nuclear accident scenarios.** While adverse health effects from exposure to ionizing radiation (termed *radiological health effects*) have understandably been the primary public health concern since the emergence of nuclear power technology, our understanding of the realistic public health consequences from nuclear accident scenarios has evolved through: (1) the accumulation of operational experience over several decades of NPP operations, including lessons learned from real-world nuclear accidents; (2) decades of research involving severe nuclear accident phenomena and NPP systems performance under severe accident conditions; and (3) the development of advanced modeling and analytical tools.

   The dominant public health consequences arising from the severe nuclear accidents at Three Mile Island, Chernobyl, and Fukushima have not been radiological health effects attributable to direct exposure to radiation released during these accidents. Rather, the dominant public health consequences arising from these accidents have been indirect effects caused in part by implementation of protective actions that aim to avoid or limit radiological dose. Notable examples of these indirect effects include psychological health effects and socioeconomic disruption (collectively termed *psychosocial effects*). While there were about 30 early fatalities among emergency response workers that were attributed to acute accidental radiation exposures during the Chernobyl
accident, no members of the general public have died from acute radiation exposures attributed to these severe nuclear accidents. Moreover, apart from detectable increases in the incidence of thyroid cancer cases among young people who were exposed to radiological materials from the Chernobyl accident, several health studies indicate that the excess cancer risk attributable to radiation exposures following these severe nuclear accidents among affected populations is negligible with respect to their background cancer risk. These findings are further bolstered by state-of-the-art PCA studies that leveraged decades of severe accident research and advanced analytical tools to improve our state of knowledge about severe accident phenomena, accidental radiological releases, and offsite radiological consequences. These contemporary studies indicate that severe nuclear accident scenarios: (1) progress more slowly; (2) release less radiological materials to the environment; and (3) result in fewer adverse radiological health effects than previous studies had indicated.

2. The QHOs do not account for risk-risk tradeoffs between radiological and non-radiological health risks. A related issue is that the QHOs do not account for tradeoffs between radiological health risks and non-radiological health risks attributed to offsite protective actions taken to avert radiological dose to the affected population. One reason the early fatality and latent cancer fatality risks attributable to nuclear accident scenarios are relatively low is

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1 In this context, the term negligible means that the increase in cancer risk attributable to accidental radiation exposures is not large enough to be statistically discernible from normal variation in baseline cancer rates among the affected population.
that—in both modeled and real-world accident scenarios—protective actions such as sheltering, evacuation, and dose-dependent relocation are implemented to avoid or limit accidental radiation exposures among the affected population. However, implementation of such protective actions exposes people to other health and safety risks, including: (1) the risk of injury or death during evacuation; and (2) the risks of psychosocial effects from permanent or long-term relocation. Implementation of offsite protective actions to avert radiological dose thus involves a tradeoff between radiological and non-radiological health risks. Yet the QHOs cannot account for such risk-risk tradeoffs, and therefore portray an incomplete picture of the public health risks attributable to nuclear accident scenarios and related protective action decisions that aim to mitigate radiological health risk.22

3. **The QHOs are not able to distinguish between nuclear accident scenarios involving different levels of severity with respect to societal radiological risks.** Another issue with the QHOs arises from the averaging technique that is used to estimate average individual early fatality risk and average individual latent cancer fatality risk for comparison with the QHOs. In practice, these average individual risk measures are calculated by summing the total number of radiological health effect cases predicted to occur within the specified region and then normalizing this result by dividing by the total number of people in the exposed population within that region. Using this averaging technique, a hypothetical accident that causes 100 early deaths among an exposed population of 10,000,000 people results in the same average individual fatality risk (1E-05) as a hypothetical accident that causes 1 early death among
an exposed population of 100,000 people. As a result, since the QHOs do not include measures of societal risk that reflect the total numbers of adverse radiological health effects, they are unable to distinguish between nuclear accident scenarios that involve different levels of severity with respect to societal radiological risks.  

4. The QHOs may not be sensitive to multi-unit accident effects. Section 1.C.1 described the need to account for the risk contribution from multi-unit accident scenarios in safety goal policy applications. Yet the QHOs may not be sensitive to the effects of including the risk contribution from multi-unit accident scenarios for two reasons: (1) NPP PRA models credit offsite protective actions and typically assume most of the population living in the vicinity of (or area near) an NPP site will begin orderly evacuation when it appears any reactor unit at the site may experience damage to the nuclear fuel in the reactor core (hereafter core damage); and (2) as stated above, the QHOs do not account for risk-risk tradeoffs between radiological and non-radiological health risks. For multi-unit accident scenarios, this means that most of the nearby population would likely be evacuated by the time any radiological releases from additional reactor units would occur, resulting in little (if any) effect on average individual early fatality risk. In addition, while multi-unit accident scenarios would be expected to result in larger radiological releases to the environment, affected populations would likely be relocated to avert radiological dose, thus limiting the effect of such scenarios on average individual latent cancer fatality risk. Therefore, although it is expected that multi-unit accident scenarios would result in contamination of larger land areas and/or contamination at higher levels of
radioactivity—thereby resulting in the need to evacuate or relocate larger numbers of people relative to single-unit accident scenarios—the QHOs are not able to account for these effects, and thus may not be sensitive to multi-unit accident effects.

Taken together, these four issues that arise from the limited scope of the QHOs underscore the need for considering a broader set of public health risk metrics in safety goal policy applications. Using the existing QHOs as the basis for safety goal screening evaluations, proposed actions designed to reduce societal risks of radiological health and non-radiological health consequences may be rejected before detailed cost-benefit analyses would be performed to determine whether the actions could result in a net benefit to society and thus improve societal welfare. In principle, expanding the safety goal policy to include consideration of a broader set of public health risk metrics could result in retaining some proposed regulatory actions for detailed cost-benefit analyses that would otherwise be screened from further consideration if only the existing QHOs are used as the basis for safety goal screening evaluations.
1.D. Research Aims

The overall aim of this dissertation research is to evaluate the effects of expanding the scope and application of the safety goal policy to include consideration of: (1) the risk contribution from multi-unit accident scenarios for multi-unit NPP sites; and (2) a broader set of public health risk metrics that includes measures of societal risk for radiological health and non-radiological health consequences. From this overall research aim, three specific aims were developed to guide this investigation:

1. **Specific Aim 1: Base Case Analyses.** Specific Aim 1 is to evaluate the effect of including the contribution from multi-unit accident scenarios to selected risk metrics in safety goal policy applications. For this aim, base case analyses were performed that relied on two assumptions that respectively influence the frequency and conditional consequence elements of the risk triplet for multi-unit accident scenarios:

   a. **Multi-unit accident scenario frequency and assumed level of inter-unit dependence.** The assumed level of dependence between co-located reactor units impacts the estimated frequency of multi-unit accident scenarios. For a given NPP site and single-unit accident scenario frequency, greater levels of assumed inter-unit dependence will result in greater estimates of multi-unit accident scenario frequency. A parameter that reflects the level of inter-unit dependence for a multi-unit NPP site is the conditional probability of a multi-unit accident scenario occurring, given that a single-unit accident involving any unit at the NPP site occurs. For base case analyses, the value of this parameter was assumed to be 0.1. This implies a 10% chance of a co-located unit experiencing a concurrent accident scenario,
given that a single-unit accident scenario involving any unit occurs at the site. Results and insights from previous multi-unit NPP PRA studies and analyses of operational 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. Each multi-unit accident scenario can have a unique conditional probability given the occurrence of a specified single-unit accident scenario. However, given the absence of sufficient data to develop reliable estimates of scenario-specific conditional probabilities, a single average value was assumed to apply globally across all multi-unit accident scenarios.

b. **Multi-unit accident scenario conditional consequences and assumed timing offset between concurrent accident scenarios involving co-located reactor units.** The assumed timing offset (delay time) between concurrent accident scenarios involving multiple reactor units co-located at a shared NPP site impacts the timing of radiological releases from co-located reactor units and thus the conditional consequences of multi-unit accident scenarios. For base case analyses, multi-unit accident scenarios were assumed to occur simultaneously, with no timing offset between concurrent accidents involving co-located reactor units. The hypothesis underlying this assumption is that simultaneous accidents will result in more severe offsite consequences, though this may not be the case for all consequence metrics of interest.
2. **Specific Aim 2: One-Way Sensitivity Analyses for Level of Inter-Unit Dependence.** Specific Aim 2 is to evaluate the effect on findings from Specific Aim 1a 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 over a range of plausible values the conditional probability of an accident in a co-located unit, given that an accident involving at least one unit at the NPP site occurs.

3. **Specific Aim 3: One-Way Sensitivity Analyses for Timing Offset.** Specific Aim 3 is to evaluate the effect on findings from Specific Aim 1b 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.
1.E. Policy Analysis Design

1.E.1. Safety Goal Policy Alternatives

Two safety goal policy alternatives were selected to evaluate the effects of expanding the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics:\footnote{A third option that also represents a hypothetical expansion in scope of safety goal policy is also possible. In addition to including contributions from both single-unit and multi-unit accidents, such an option would assume the policy is applied to the entire NPP site, rather than to individual reactor units. In addition to accidents involving one or more reactor units, such an option could include accidents involving other major sources of radiological materials at the NPP site, including facilities for storing spent nuclear fuel. However, this option is not evaluated as part of this dissertation research for two main reasons: (1) the USNRC licenses and regulates NPPs at the level of individual reactor units; and (2) since previous PRA studies have shown that accident scenarios involving operating reactor units are the dominant contributor to accident risk involving commercial NPP sites, accident scenarios involving other elements of the nuclear fuel cycle (i.e., spent nuclear fuel) are explicitly excluded from the scope of the safety goal policy.\footnote{Additional limitations in the analysis scope for this dissertation research are described in Section 1.G.}}:

1. **Option 1 (Status Quo): Safety Goal Policy Applied on a Per-Reactor-Unit Basis with Only Single-Unit Accident Scenarios Included.** This option represents the status quo with respect to application of the safety goal policy. For this option, the policy is applied to individual reactor units, rather than to the entire NPP site, and only the contribution from single-unit accident scenarios is included in estimating selected risk metrics.

2. **Option 2 (Hypothetical Expansion): Safety Goal Policy Applied on a Per-Reactor-Unit Basis with Both Single-Unit and Multi-Unit Accident Scenarios Included.** This option represents a hypothetical expansion in scope and application of the safety goal policy. Although the policy is still applied to individual reactor units, as with the status quo option, the contributions from
both single-unit accident scenarios and multi-unit accident scenarios are included in estimating selected risk metrics for each reactor unit.

1.E.2. Risk Metrics and Figure of Merit

The risk metrics selected to evaluate the effects of expanding the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics are summarized below in Table II. This set of metrics is intended to represent three broad public health risk perspectives: (1) individual radiological health risk; (2) societal radiological health risk; and (3) societal non-radiological health risk. The basis for each spatial interval or region over which each selected risk metric is calculated is provided in Footnotes a and b of Table II.

As noted in Footnote c of Table II, the total numbers of people relocated during the emergency and late (recovery) phases of accident response represent indirect, surrogate measures for adverse non-radiological health effects attributed to protective actions taken to avert radiological dose among the affected population. There are three principal reasons for selecting these indirect measures in lieu of direct measures of adverse non-radiological health effects: (1) estimates of these indirect measures can be calculated using available technology; (2) these indirect measures

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a U.S. Environmental Protection Agency (USEPA) guidance identifies and defines three radiological incident phases: (1) early phase (hours to days): the beginning of a radiological incident when immediate decisions for effective use of protective actions are required and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions; (2) intermediate phase (weeks to months): the period beginning after the radiological source and releases have been brought under control (has not necessarily stopped but is no longer growing) and reliable environmental measurements are available for use as a basis for decisions on protective actions and extending until these additional protective actions are no longer needed; and (3) late phase (months to years): the period beginning when recovery actions designed to reduce radiation levels in the environment to acceptable levels are commenced and ending when all recovery actions have been completed.
measures can provide insights into a range of potential adverse non-radiological health effects attributable to implementing protective actions, rather than limiting insights to a select set for which limited data are available; and (3) results from a recent study suggest that the number of people relocated is a good and relatively straightforward to calculate proxy measure for societal disruption caused by potential nuclear accident scenarios.22

Table II. Risk Metrics Selected to Achieve Research Aims

<table>
<thead>
<tr>
<th>Risk Perspective</th>
<th>Selected Risk Metric</th>
<th>Spatial Interval</th>
</tr>
</thead>
<tbody>
<tr>
<td>Individual Radiological Health Risk</td>
<td>Average Individual Early Fatality Risk</td>
<td>0-1 mile(^a)</td>
</tr>
<tr>
<td></td>
<td>Average Individual Latent Cancer Fatality Risk</td>
<td>0-10 miles(^a)</td>
</tr>
<tr>
<td>Societal Radiological Health Risk</td>
<td>Total Number of Early Fatality Cases</td>
<td>0-50 miles(^b)</td>
</tr>
<tr>
<td></td>
<td>Total Number of Latent Cancer Fatality Cases</td>
<td>0-50 miles(^b)</td>
</tr>
<tr>
<td>Societal Non-Radiological Health Risk</td>
<td>Total Population Relocated During Emergency Phase(^c)</td>
<td>0-50 miles(^b)</td>
</tr>
<tr>
<td></td>
<td>Total Population Relocated During Late (Recovery) Phase(^c)</td>
<td>0-50 miles(^b)</td>
</tr>
</tbody>
</table>

\(^a\) The 0-1 mile and 0-10 miles spatial intervals were selected for average individual health risk metrics to be consistent with the region defined for each QHO specified in the USNRC safety goal policy statement.

\(^b\) The 0-50 miles spatial interval was selected for societal health risk metrics to be consistent with USNRC guidance for estimating societal consequences as part of regulatory or environmental impact analyses.

\(^c\) The total numbers of people relocated during the emergency and late (recovery) phases of accident response represent indirect, surrogate measures for adverse non-radiological health effects attributed to protective actions taken to avert radiological dose among the affected population.

One figure of merit\(^o\) (FOM) was selected to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. This FOM is the relative contribution of multi-unit accident scenarios to the total mean value for

\(^o\) A figure of merit is defined as "the quantitative value, obtained from a PRA analysis, used to evaluate the results of an application." For this dissertation research, the application is an evaluation of the effects of expanding the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics.
each selected risk metric calculated under Option 2, which assumes the risk contributions from both single-unit and multi-unit accident scenarios are included.

In addition to this FOM, the margin to each safety goal QHO\(^p\) (the early fatality risk QHO and the latent cancer fatality risk QHO) is estimated for each policy alternative. This information is used to provide supplementary insights about the effect of this expansion in the scope of the safety goal policy that are specific to the individual radiological health risk perspective, considering the existing QHOs.

1.E.3. Study Population and Accident Scenarios

Two representative U.S. NPP sites, each comprised of two co-located reactor units, were selected as case studies for this dissertation research: (1) Peach Bottom Atomic Power Station (hereafter Peach Bottom); and (2) Surry Power Station (hereafter Surry). These sites were selected because they have been the subjects of many previous PRA and PCA studies, and therefore have a vast amount of peer-reviewed information available. Peach Bottom is representative of NPP sites using the boiling-water reactor (BWR)–Mark I containment design—like those at the FDNPS in Japan—and includes Unit 2 and Unit 3.\(^q\) It is located near Lancaster, PA and has a below-average offsite population density within 10 miles of the site boundary. Surry is representative of NPP sites using the pressurized-water reactor (PWR)—large dry containment design, and includes Unit 1 and Unit 2. It is located near

\(^q\) Peach Bottom Unit 1 completed decommissioning in 1978 and is no longer operational.
Newport News, VA and has an average offsite population density within 10 miles of the site boundary.

This dissertation 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.” 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 important. The single-unit accident scenarios were selected based on PRA model results and expert judgments about their relative importance with respect to: (1) the likelihood of causing core damage; or (2) the potential to cause significant offsite radiological health consequences due to early failure or bypass of the containment structure, assuming each accident scenario were to occur. Summary information for the single-unit accident scenarios evaluated for each NPP site in the SOARCA pilot study is provided in Table III and Table IV. Table III provides information about: (1) mean accident scenario frequency; (2) times to core damage and radiological release; and (3) release fractions in terms of the percentage of core inventory for radionuclides that are important determinants of early fatality risk and latent cancer fatality risk. Table IV provides high-level descriptions about the event sequences associated with each of the single-unit accident scenarios.

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* Multi-unit accident scenarios involving concurrent accidents in co-located reactor units at each NPP site were determined to be beyond the scope of the SOARCA pilot study. For the pilot study, detailed offsite radiological consequence models were developed for single-unit accident scenarios involving Peach Bottom Unit 2 and Surry Unit 1.
To address the aims for this dissertation research, state-of-the-art consequence models were constructed for all possible inter-unit combinations of two-unit accident scenarios that could be created by combining the single-unit accident scenarios that were modeled for each NPP site in the SOARCA pilot study, assuming the two units at each site are identical and subject to the same set of single-unit accident scenarios. These two-unit accident scenario models were developed and evaluated using the MELCOR Accident Consequence 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 avert radiological dose; and (5) dose-response models used to estimate numbers of radiological health effects. MACCS was recently enhanced to include a multi-source model that enables 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 outcomes that can arise from combining sequential events and/or decisions. Figures 2 and 3 respectively illustrate: (1) nine two-unit accident scenario models for Peach Bottom Unit 2 and Unit 3 that can be created by combining the three single-unit accident scenarios evaluated for Peach Bottom as part of the SOARCA pilot study, assuming the Unit 2 accident scenario occurs first; and (2) 16 two-unit accident scenario models for Surry Unit 1 and Unit 2 that can be created by combining the four single-
unit accident scenarios evaluated for Surry, assuming the Unit 1 accident scenario occurs first. Since combinations of two-unit accident scenarios in which an accident scenario in the second unit for each NPP site (Unit 3 for Peach Bottom or Unit 2 for Surry) occurs first are also possible, the total number of two-unit accident scenarios for each NPP site is twice the number illustrated (18 for Peach Bottom and 32 for Surry). In principle, the order in which the accident scenarios involving two units occur could matter, especially if there are important differences between the two units. Therefore, it would be more appropriate to consider the permutations (ordered combinations) of two-unit accident scenarios that could occur. However, since the two units at each NPP site are assumed to be identical, these additional two-unit accident scenarios that are not illustrated in Figures 2 and 3 would result in identical conditional consequences. This assumption of identical units—which is reasonable for both NPP sites—thus eliminates the need to develop unique consequence models for these additional two-unit accident scenarios and reduces the total number of necessary two-unit consequence models by a factor of two.\(^*\)

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 selected risk metrics. One simulation represented the base case analysis that assumed the constituent accident

\(^*\) Although the assumption of identical units reduces the total number of two-unit accident scenario models needed for each NPP site by a factor of two, this does not eliminate the need to account for the contribution of the other half of two-unit accident scenarios to the frequency of two-unit accident scenarios. This issue is further addressed in Chapter 3.
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 (delay time) between concurrent accidents involving both units was varied from 1 to 7 days, in one-day increments. 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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1 A limited set of trial sensitivity analyses were also performed to evaluate the effect of varying the timing offset parameter from 1 hour to 24 hours in one-hour increments. Only results for one-way sensitivity analyses using the 1-day to 7-day range in one-day increments are presented for three reasons: (1) the 7-day range was judged to be more appropriate for the expected termination of major releases for multi-unit accident scenarios; (2) 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 (3) one-day increments appeared to provide sufficient resolution to characterize patterns or trends.
Table III. Single-Unit Accident Scenarios Evaluated in the SOARCA Pilot Study

<table>
<thead>
<tr>
<th>Single-Unit Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Time to Core Damage (hours)</th>
<th>Time to Radiological Release (hours)</th>
<th>I-131 &amp; Cs-137 Release Fractions&lt;sup&gt;a&lt;/sup&gt; (% core inventory)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Peach Bottom Atomic Power Station (Peach Bottom)</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
| Long-Term Station Blackout (LTSBO) | 3.0E-06 | 9 | 20 | I-131: 2%  
Cs-137: <1% |
| Short-Term Station Blackout – Base Case (STSBO-Base) | 3.0E-07 | 1 | 8 | I-131: 12%  
Cs-137: 2% |
| Short-Term Station Blackout with Reactor Core Isolation Cooling System Blackstart (STSBO-RCIC) | 3.0E-07 | 7 | 17 | I-131: 2%  
Cs-137: <1% |
| **Surry Power Station (Surry)** | | | | |
| Long-Term Station Blackout (LTSBO) | 2.0E-05 | 16 | 45 | I-131: <1%  
Cs-137: <1% |
| Short-Term Station Blackout – Base Case (STSBO-Base) | 2.0E-06 | 3 | 25 | I-131: 1%  
Cs-137: <1% |
| Short-Term Station Blackout with Thermally-Induced Steam Generator Tube Rupture (STSBO-TISGTR) | 4.0E-07 | 3 | 3.5 | I-131: 1%  
Cs-137: <1% |
| Interfacing Systems Loss-of-Coolant Accident (ISLOCA) | 3.0E-08 | 13 | 13 | I-131: 16%  
Cs-137: 2% |

<sup>a</sup> Iodine-131 (I-131) is generally representative of radionuclides with shorter half-lives and provides a surrogate measure of early fatality risk. Cesium-137 (Cs-137) is generally representative of radionuclides with longer half-lives and provides a surrogate measure of latent cancer fatality risk.
<table>
<thead>
<tr>
<th>Single-Unit Accident Scenario</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>Station Blackout (SBO)</td>
<td>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.</td>
</tr>
<tr>
<td>Long-Term Station Blackout (LTSBO)</td>
<td>An earthquake causes a loss of all AC power sources, but onsite batteries can supply DC power to safety systems for about 4-8 hours until battery depletion.</td>
</tr>
<tr>
<td>Short-Term Station Blackout – Base Case (STSBO-Base)</td>
<td>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. Thus, onset of damage to nuclear fuel in the reactor core occurs in the “short-term.” This is the base case STSBO.</td>
</tr>
<tr>
<td>Short-Term Station Blackout with Reactor Core Isolation Cooling System Blackstart (STSBO-RCIC)</td>
<td>This scenario 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.</td>
</tr>
<tr>
<td>Short-Term Station Blackout with Thermally-Induced Steam Generator Tube Rupture (STSBO-TISGTR)</td>
<td>This scenario 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.</td>
</tr>
<tr>
<td>Interfacing Systems Loss-Of-Coolant Accident (ISLOCA)</td>
<td>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.</td>
</tr>
</tbody>
</table>
Figure 2. Two-Unit Accident Scenario Models for Peach Bottom Unit 2 and Unit 3. Nine two-unit accident scenario models were constructed by combining the three single-unit accident scenario models for Peach Bottom Unit 2 that were evaluated in the SOARCA pilot study.
Sixteen two-unit accident scenario models were constructed by combining the four single-unit accident scenario models for Surry Unit 1 that were evaluated in the SOARCA pilot study.
1.F. Key Assumptions

The study design for this dissertation research relies on five key assumptions that could threaten the validity of its research findings:

1. **The consequence models from the SOARCA pilot study are assumed to be 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 dissertation 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.\(^{16-18}\) However, any bias introduced by modeling choices made in the SOARCA pilot study would necessarily carry over into this dissertation research and may bias its findings.

2. **The two operating reactor units co-located at each NPP site are assumed to be 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,\(^u\) while the second represents the concurrent accident scenario that occurs in the co-located unit. Since all the models for the SOARCA pilot study were based on

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\(^u\) The need for identifying which unit serves as the reference unit for the two-unit accident scenarios is explained next with Key Assumption #3.
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 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 that give rise to subtle differences in their risk profiles.

3. **One unit is assumed to always serve 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 study’s 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 accident scenarios modeled and evaluated are assumed to be representative of the full spectrum of potential accident scenarios for each NPP site.** The safety goal QHOs were developed for comparison with results for corresponding risk 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 and a broader set of public health risk metrics, 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 selected risk metrics. As described in Chapter 3, this assumption allows for calibrating the results of each representative accident scenario to account for the contribution to frequency from those scenarios belonging to the representative category that are not modeled, so that a more accurate estimate of total accident risk can be developed. However, this assumption can lead to biased findings 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 selected risk metrics.
5. The modeled NPP sites are assumed to be representative of the population of multi-unit NPP sites. The two NPP sites selected for modeling and evaluation as part of this dissertation 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 like 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. However, this assumption can lead to biased generalizations if attempting to apply the findings from this dissertation research to multi-unit NPP sites that differ in important ways from the study population. Examples of differences that could result in biased generalizations include differences in: (1) statistical variability in offsite weather conditions over time; (2) offsite population density; and (3) offsite emergency response plans and criteria for implementing protective actions that aim to avert radiological dose.
1.G. Analysis Scope

As stated in Section 1.F.4, the safety goal QHOs were developed for comparison with results for corresponding risk 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. A full-scope NPP Level 3 PRA can include accident scenarios that: (1) are initiated by hazards that are either internal or external to the NPP; (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).

This dissertation 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 unique two-unit accident scenarios illustrated in Figures 2 and 3 that could be constructed by combining these single-unit accident scenarios, assuming the two units at each NPP site are identical. Thus, 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 excluded from full-scope NPP Level 3 PRAs because a probabilistic treatment of such scenarios is
considered to be beyond the state-of-the-art. Moreover, such scenarios are explicitly excluded from the scope of the safety goal policy for this reason.1

2. **Reactor accident scenarios that occur while the reactor is in plant operating states other than at-power.** These include reactor accident scenarios that could occur while the reactor is in low-power or shutdown plant operating states. All single-unit accident scenarios that were modeled and evaluated as part of the SOARCA pilot study were assumed to occur while the operating reactor units were operating at full power. Although reactor accident scenarios could occur while the reactor is operating in low-power or shutdown modes, these scenarios have been shown in previous PRA studies to be minor risk contributors compared to those that could occur while the reactor is operating at full power.

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. All single-unit accident scenarios that were modeled and evaluated as part of the SOARCA pilot study were for operating reactor units, which previous PRA studies have shown to be the dominant contributor to accident risk involving commercial NPP sites.19 Moreover, such scenarios are also explicitly excluded from the scope of the safety goal policy for this reason.1

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v Although accident scenarios initiated by deliberate malevolent acts are not addressed in full-scope NPP PRAs, the USNRC considers such scenarios in its set of deterministic design-basis threats (DBTs) that are specified in its security and physical protection regulations. These regulations require NPP licensees to demonstrate they can defend against the DBTs with a high level of assurance.27
1.H. Significance of Research and Potential Policy Implications

This dissertation research makes three significant contributions to the literature:

1. It specifies and applies efficient models for estimating the contributions to selected risk metrics from categories of single-unit and multi-unit accident scenarios using state-of-the-art consequence models from a contemporary PCA study that leveraged decades of severe accident research and advanced analytical tools to develop realistic estimates of the offsite radiological consequences attributable to important nuclear accident scenarios. These models were demonstrated for the two-unit case using two NPP sites that are representative of a broad category 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 accident scenarios involving multiple co-located reactor units; and (2) staggered accident scenarios in which the timing offset (delay time) 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 selected risk metrics; and (2) the effects of expanding the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics that portray a more complete picture of the public health risks attributable to potential single-unit and multi-unit accident scenarios.
Insights derived from this dissertation 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: (1) consideration of the risk contribution from multi-unit accident scenarios; and (2) QHOs that address societal risk. 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. This could have significant implications for regulatory requirements, policies, or guidance pertaining to defense-in-depth elements designed to limit the public health risks attributable to potential accident scenarios involving operating reactor units at multi-unit NPP sites. Three 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 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.\textsuperscript{30} 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; 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.\textsuperscript{31} 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.
1.I. Chapter Summary and Dissertation Overview

The USNRC safety goal policy is used to determine whether proposed regulatory actions that aim to enhance NPP safety beyond the level deemed necessary for ensuring the public is adequately protected would provide a substantial enough increase in public protection relative to the existing level of residual risk to warrant further evaluation using detailed cost-benefit analyses. Under this policy, a limited set of results from NPP PRAs are compared against corresponding quantitative objectives to determine whether and to what extent qualitative safety goals have been attained.

This chapter introduced central concepts relevant to the safety goal policy and probabilistic analysis techniques used to measure attainment of safety goals, as well as the limitations in the scope of the safety goal policy that motivate this dissertation research. It identified two needs with respect to the scope and application of the safety goal policy: (1) the need to account for the risk contribution from multi-unit accident scenarios; and (2) the need for including a broader set of public health risk metrics that address the societal risk of radiological and non-radiological health consequences, not just individual risk of experiencing radiological health effects. This dissertation research aims to evaluate the effects of a hypothetical expansion in the scope of the safety goal policy to address both needs.

This chapter further describes principal design aspects of a policy analysis that has been performed to achieve this overall research aim, including: (1) safety goal policy alternatives; (2) risk metrics and FOM; (3) study population and accident scenarios; (4) key assumptions; and (5) analysis scope.
This dissertation is organized into five chapters. A brief overview of the topics addressed in each of the remaining chapters follows.

Chapter 2 represents the product of a literature review and provides a more detailed description of the background information that is germane to this dissertation research than what was introduced in Chapter 1. Topics addressed in Chapter 2 include: (1) development, application, and limitations of the USNRC safety goal policy; (2) probabilistic analysis techniques for NPPs, with an emphasis on the traditional scenario-based approach to PRA; (3) insights about multi-unit accident scenarios derived from previous PRAs, operational experience, and the 2011 Fukushima nuclear accident; (4) the evolution of our state of knowledge about the public health risks attributable to nuclear accident scenarios and the need for including a broader set of public health risk metrics in safety goal policy applications; and (5) the objectives, design, and key conclusions from the SOARCA pilot study, which leveraged decades of severe accident research and advances in analytical tools to develop the state-of-the-art consequence models that this dissertation research builds upon.

Chapter 3 describes the models and analytical tools used to perform the policy analysis and to address the specific aims for this dissertation research. The chapter explains the need for efficient risk estimation models that can be used to calibrate the frequency and conditional consequence results based on SOARCA models using results from previous full-scope NPP PRA studies. This calibration is designed to estimate approximately equivalent risk results that can then be used to extract meaningful safety goal policy insights, without having to perform a resource-intensive, contemporary full-scope NPP PRA. Analytical tools are summarized, with
a detailed treatment of the MACCS suite of analytical tools that is used to perform PCA. The chapter delineates and illustrates the process and equations used to estimate: (1) the contribution from single-unit accident scenarios to selected risk metrics; (2) the contribution from two-unit accident scenarios to selected risk metrics; (3) the values of selected risk metrics for each policy alternative; and (4) the values of the FOM used to evaluate the effect of expanding the scope of the safety goal policy to include multi-unit accident scenarios and a broader set of public health risk metrics. Additional consideration is given to how these methods—which can be directly applied to NPP sites comprised of two co-located operating reactor units—can be generalized for application to NPP sites comprised of more than two units.

Chapter 4 presents the results of the policy analysis and its policy implications. Results for selected risk metrics are summarized for both single-unit and two-unit accident scenarios under base case analysis assumptions. Results for the FOM used to evaluate the effect of the hypothesized expansion in the scope of the safety goal policy are presented and interpreted for three different risk perspectives represented by the selected risk metrics: (1) individual radiological health risk perspective; (2) societal radiological health risk perspective; and (3) societal non-radiological health risk perspective. Finally, results and insights from one-way sensitivity analyses designed to evaluate the effect of variation in two factors are presented: (1) the assumed level of dependence between co-located units, which impacts the frequency of modeled two-unit accident scenarios; and (2) the assumed timing offset (delay time) between concurrent accident scenarios involving co-located reactor units, which impacts the conditional consequences of modeled two-unit accident scenarios.
Finally, Chapter 5 describes the conclusions and recommendations developed based on this dissertation research. The chapter addresses: (1) what has been accomplished; (2) key conclusions derived from interpretation of results; (3) reflections on the value added by applying a broader public health perspective to USNRC risk management or safety-related decisions involving NPPs; and (4) recommendations for further work, including additional research to address known limitations of this dissertation research.
Chapter 2. Background

2.A. Chapter Introduction and Overview

Chapter 1 introduced central concepts relevant to the U.S. Nuclear Regulatory Commission (USNRC) safety goal policy\(^1\) and probabilistic analysis techniques used to measure attainment of safety goals, as well as the limitations in the scope of the safety goal policy that motivate this research. It further described principal design aspects of a policy analysis that has been performed to evaluate the effects of a hypothetical expansion in the scope of the safety goal policy to address these limitations.

This chapter represents the product of a literature review and provides a more detailed description of the background information that is germane to this research than what was introduced in Chapter 1.
2.B. U.S. Nuclear Regulatory Commission Safety Goal Policy Statement

2.B.1. Introduction

A question that emerged in the late 1970s and early 1980s as the USNRC and nuclear industry were responding to the March 1979 accident involving Unit 2 of the Three Mile Island Nuclear Station (hereafter Three Mile Island) was “How safe is safe enough?” with respect to imposing additional regulatory requirements designed to further enhance the safety of commercial nuclear power plants (NPPs) beyond the level needed to ensure the public is adequately protected. The USNRC’s approach to addressing this question was to develop a safety goal policy that broadly defines an acceptable level of radiological risk to the public from NPP operations. 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, the background discussion provided here is relatively brief and focuses on the essential issues that are germane to this research.

The USNRC derives its statutory responsibilities and authorities from the Atomic Energy Act of 1954, as amended (AEA). 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 from the hazards posed by the nuclear technologies it regulates in executing its statutory functions.

Federal courts have ruled in relevant case law that achieving a state of adequate protection is not equivalent to achieving a state of zero risk, which could only be achieved if nuclear technologies were eliminated altogether; such an option is not feasible, given our society’s reliance on nuclear power as a major source of electricity, and the long-lived nature of radioactive waste that must continue to be managed. Achieving adequate protection thus means NPPs must pose no undue risk (not zero risk) to public health and safety. As a result, the public is still exposed to some level of risk that remains when the USNRC has determined that an NPP has achieved a state of adequate protection; this remaining risk is termed residual risk.

* 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 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.
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.” These provisions: (1) suggest that Congress 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 its legislative history to determine Congressional intent in establishing the USNRC’s statutory mandate.

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. While no explicit definition of what constitutes adequate protection exists, an NPP that complies with the set of relevant USNRC regulatory requirements is presumed to be in an adequate protection state. However, if an NPP is unable to comply with one or more USNRC regulatory requirements, this does not necessarily mean that the public is not adequately protected; the USNRC has sufficient flexibility to grant exemptions to its regulatory requirements on a case-specific basis, provided it determines that an adequate protection state can still be achieved with such exemptions in place.
2.B.3. Safety Goal Policy Development

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 at Three Mile Island; 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 considering 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. 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.

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; and (3) a two-year trial-use and evaluation period that resulted in some substantive policy changes before final publication and implementation.

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* 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.
2.B.4. Safety Goal Policy Application

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 reactor units at commercial NPP sites. The safety goal policy represents a key aspect of the USNRC’s risk management philosophy. It represents a value judgment that essentially addresses the question of “How safe is safe enough?” for regulatory decisions regarding NPP safety. This policy is a product of an agency effort to implement control mechanisms designed to avoid imposing costly regulatory requirements that may not be warranted relative to their potential safety benefit and the level of residual risk to the public. In practice, it guides agency screening evaluations to determine whether proposed regulatory actions that would impose additional generic requirements to enhance NPP safety beyond those needed to ensure adequate protection could provide a substantial enough safety benefit—relative to the level of residual risk to the public—to warrant further evaluation. Using this approach, proposed regulatory actions that aim to further enhance NPP safety can be rejected before performing detailed cost-benefit analyses to determine whether they could be justified based on their net present value (NPV).

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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 demonstrate conformance with the safety goal policy. This research therefore focuses only on the application of USNRC safety goals in the context of public health risks arising from potential accidental releases of radiological materials from operating power reactors at commercial NPPs.
In principle, rejection of a proposed regulatory action would occur if either: (1) the potential safety benefit is judged to be not substantial enough; or (2) the residual risk to the public is determined to be at an acceptably low level, and thus limited resources that would be applied to the proposed regulatory action could be better applied to alternative courses of action. The concept of opportunity cost from microeconomic theory supports this reasoning. Since society has finite resources to expend on safety enhancements, excessive spending to reduce the health and safety risks posed by commercial NPP operations could potentially increase net public risk by diverting scarce safety resources from more effective risk reduction activities.

The safety goal policy is based on a hierarchical framework comprised of two high-level qualitative safety goals that are each supported by two lower-level quantitative health objectives (QHOs). These QHOs 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 fatality risk (hereafter 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 latent cancer fatality risk QHO). Figure 4 illustrates this hierarchical framework, while Table I in Chapter 1 provides the exact language used to specify the qualitative safety goals and supporting QHOs in the USNRC Safety Goal Policy Statement.
Figure 4. 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 attainment of each qualitative safety goal. 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 NPV of net benefits (termed net value), which is the difference between the sum of monetized and discounted benefits and the sum of monetized and discounted costs.²

However, results from NPP probabilistic risk analysis (PRA) or probabilistic consequence analysis (PCA) studies 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 such detailed cost-benefit analyses would be performed. USNRC regulatory analysis guidelines include guidance for performing a screening evaluation of proposed regulatory actions with respect to the USNRC safety goals. This safety goal screening evaluation is designed to identify in part when a regulatory requirement should not be imposed generically on NPPs because the residual risk to the public is determined to be at an acceptably low level. In this way, it is intended to eliminate some proposed regulatory actions from further consideration, regardless of whether they could be justified based on their net value. This safety goal screening evaluation can also be used to determine whether a proposed generic safety enhancement backfit that does not meet certain exemption criteria provides a substantial enough 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 based on their net value.\(^2\)

USNRC regulatory analysis guidelines include explicit safety goal screening criteria related to: (1) changes in the frequency of accidents involving onset of damage to nuclear fuel in the reactor core (hereafter *core damage*), and (2) conditional containment failure probabilities, conditioned on the assumed occurrence of a core damage accident. These criteria—which are intended to provide a balanced consideration of measures to prevent and mitigate core damage accidents—can be used to evaluate intermediate results from NPP PRAs to determine conformity with

\(^2\) Three exemption criteria are explicitly specified in the USNRC regulation that addresses backfitting: (1) regulatory action is necessary to bring a facility into compliance with USNRC regulatory requirements, or into conformance with written commitments made by the licensee; (2) regulatory action is necessary to ensure that the facility provides adequate protection to public health and safety; or (3) regulatory action involves defining or redefining what level of protection to public health and safety should be regarded as adequate.\(^3\)
subsidiary quantitative objectives based on core damage frequency (CDF) and large early release frequency (LERF).\textsuperscript{aa,bb} 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 PRAs for these metrics can be used to evaluate proposed regulatory actions with respect to these QHOs.\textsuperscript{2}

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 an NPP 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 through implementation of the proposed regulatory action; and (2) averted economic costs, including offsite property damage caused by contamination with radiological materials that is averted through implementation of the proposed regulatory action.\textsuperscript{2}

\textsuperscript{aa} LERF is the frequency of rapid, unmitigated releases of airborne radioactivity from the containment to the environment that occurs before effective implementation of offsite protective actions, such that there is a potential for early radiological health effects (i.e., injuries or fatalities) among the population living in the vicinity of an NPP site.\textsuperscript{40}

\textsuperscript{bb} The USNRC has determined that a quantitative objective of CDF < 1E-04 per year is a useful surrogate for the latent cancer fatality risk QHO and that a quantitative objective of LERF < 1E-05 per year is a useful surrogate for the early fatality risk QHO. However, it is important to note that the previous PRA studies that were used to evaluate the relationship between these subsidiary quantitative objectives and the QHOs did not include consideration of multi-unit accident scenarios.\textsuperscript{41}
2.B.5. Limitations in the Scope and Application of the Safety Goal Policy

2.B.5.I. The Need to Account for Risk Contributions from Multi-Unit Accident Scenarios in Safety Goal Policy Applications

During development and evaluation of the safety goal policy, the USNRC decided that the safety goals and corresponding QHOs would be applied strictly on a per-reactor-unit basis, even for the nearly 75% of U.S. reactors located at multi-unit NPP sites that include multiple co-located reactor units. 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. As a result, multi-unit accident scenarios have—with few exceptions—traditionally been excluded from NPP PRAs and safety goal screening evaluations in support of analyses of proposed regulatory actions.

Section 1.C.1. of Chapter 1 further addresses this issue and identifies three compelling reasons for expanding the scope and application of the safety goal policy to include multi-unit accident scenarios: (1) findings from previous NPP PRAs that included a limited treatment of multi-unit accident scenarios suggest that the contribution to reactor accident risk from multi-unit accident scenarios is not negligible and could be significant; (2) operational experience demonstrates that adverse events (e.g., abnormal occurrences, incidents, or accidents) involving

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Application of the safety goal policy on a per-reactor-unit basis means the policy is applied to individual reactor units at an NPP site, rather than collectively applying the policy to all reactor units at an NPP site. Moreover, single-unit NPP PRAs that provide input to safety goal screening evaluations for individual reactor units are performed under the key assumption that any reactor units co-located with the reactor unit of interest are in a safe and stable condition, thereby excluding multi-unit accident scenarios from the scope of the analysis.
multiple operating reactor units co-located at a shared site occur at a non-negligible frequency; and (3) logical and ethical reasoning show that people living in the vicinity of (or area near) an NPP are exposed to the health and safety risks posed by both single-unit and multi-unit accident scenarios.

Exclusion of multi-unit accident scenarios from the scope and application of the safety goal policy can have important implications. Since the risk contribution from multi-unit accident scenarios is not included in 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 screening 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 inappropriately conclude that these proposed actions would not be justified based on a determination that: (1) the potential safety benefit is not substantial enough; or (2) the residual risk to the public is at an acceptably low level. These proposed actions would therefore be rejected before detailed cost-benefit analyses would be performed to determine whether the actions could result in a net benefit to society and thus improve societal welfare. In principle, including the contribution to residual risk from multi-unit accidents could result in retaining some proposed regulatory actions for detailed cost-benefit analyses that would otherwise be screened from further consideration if only the risk contribution from single-unit accident scenarios is included.
2.B.5.II. The Need for Including a Broader Set of Public Health Risk Metrics in Safety Goal Policy Applications

Another limitation pertains to the scope of the QHOs that are used to measure attainment of the high-level qualitative safety goals. The QHOs are limited to two objectives that address risks to average individuals in the vicinity of (or area near) NPPs of dying from acute tissue damage (or cancer) caused by radiation exposures resulting from accidental releases from these NPPs. Section 1.C.2. of Chapter 1 further addresses this limitation in scope and identifies four issues that arise from it: (1) the QHOs do not address psychosocial and other non-radiological health effects that we now know to be the dominant public health risks attributable to potential nuclear accident scenarios; (2) the QHOs do not account for risk-risk tradeoffs between radiological and non-radiological health risks; (3) the QHOs are not able to distinguish between nuclear accident scenarios involving different levels of severity with respect to societal radiological risks; and (4) although it is expected that multi-unit accident scenarios would result in more severe non-radiological consequences relative to single-unit accident scenarios due to contamination of larger land areas and/or contamination at higher levels of radioactivity (which would cause greater numbers of people to be relocated to avert radiological dose), the QHOs are not able to account for these effects, and thus may not be sensitive to multi-unit accident effects.

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\textsuperscript{dd} The evolution of our state of knowledge about the public health risks attributable to nuclear accident scenarios is addressed in Section 2.E.
Taken together, these four issues that arise from the limited scope of the QHOs underscore the need for considering a broader set of public health risk metrics in safety goal policy applications. Using the existing QHOs as the basis for safety goal screening evaluations, proposed actions designed to reduce societal risks of radiological health and non-radiological health consequences may be rejected before detailed cost-benefit analyses would be performed to determine whether the actions could result in a net benefit to society and thus improve societal welfare. In principle, expanding the safety goal policy to include consideration of a broader set of public health risk metrics could result in retaining some proposed regulatory actions for detailed cost-benefit analyses that would otherwise be screened from further consideration if only the existing QHOs are used as the basis for safety goal screening evaluations.
2.C. Probabilistic Analysis Techniques for Nuclear Power Plants

2.C.1. Overview of the Traditional Scenario-Based Approach to Probabilistic Risk Analysis and Its Uses

To effectively manage the risks 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, including commercial NPPs. 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:\(^3\):

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. There are two major groups of initiating event hazards: (1) internal hazards—hazards arising from sources within the boundaries of the system being modeled and analyzed (e.g., random failures of structures, systems, or components (SSCs), or human operator errors); and (2) external hazards—hazards arising from sources outside the system boundary (e.g., naturally occurring events, other technological accidents, or deliberate malevolent acts). An initiating event challenges mitigating systems designed to prevent adverse outcomes from occurring. Each accident scenario then ends with an adverse
outcome or end state of interest. Within each scenario, there can be multiple intermediate pivotal 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 the product of: (1) its initiating event frequency; and (2) 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 exposures to harmful agents (e.g., population doses arising from exposures to ionizing radiation); (2) adverse human health effects resulting from exposures to harmful agents (e.g., early fatalities from acute radiation exposure and latent cancer fatalities from chronic radiation exposure); (3) adverse environmental impacts; and (4) economic damages or financial loss.

The accident scenario concept is analogous to Reason’s Swiss Cheese Model of System Accidents. In this model, a complex engineered system includes multiple layers of defense or mitigating systems—engineered safety features, physical
barriers, safeguards, and/or procedures for operator actions—designed to prevent or mitigate an accident arising from events initiated by internal or external hazards. This is known in many industries, including the commercial nuclear power industry, as the “defense-in-depth” approach. In the ideal case, each of these layers of defense is independent and perfectly intact, thereby preventing an initiating event from ever leading to one or more adverse end states. However, each of these defense layers is vulnerable to failure, whether caused by: (1) active events that directly impact the integrity of the defense at the time of an accident; or (2) latent conditions—hidden defects in the design, operation, or maintenance of the system and its defense layers—that can remain dormant within the system for years until they combine with initiating events and active failures to create an accident opportunity. Within the Swiss Cheese Model of System Accidents, these vulnerabilities in defense layers are represented by holes in the defenses, with each layer of defense being modeled as a layer of Swiss cheese. Although such holes are fixed or static in an actual slice of Swiss cheese, the model represents these vulnerabilities as dynamic holes that can change over time and space, depending on the conditions under which the system is operating. The Swiss cheese model then portrays an accident opportunity as a scenario in which the holes of all successive defense layers align to provide a trajectory for an event initiated by a hazard to lead to an accident resulting in damage or loss. In the traditional scenario-based approach to PRA, there are many possible ways for the holes of all successive defense layers to align to create an accident trajectory, with each accident scenario representing one unique trajectory in which a particular initiating event can result in one or more adverse end states. The goal of a PRA is to identify and model a reasonably complete set of potential
accident scenarios that can result from events initiated by the set of internal and external hazards to which the system of interest is exposed.

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

\[ R_{TOTAL} = \{ < s_i, f_i, c_i > \} \forall i \]

- \( R_{TOTAL} \) = the total risk attributed to failure of the modeled system.
- \( i \) = index of accident scenarios.
- \( s_i \) = accident scenario \( i \).
- \( f_i \) = frequency of accident scenario \( i \).
- \( c_i \) = vector of conditional consequences, assuming accident scenario \( i \) occurs.

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

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\textsuperscript{ee} In practice, it is usually not possible to identify and explicitly model all possible accident scenarios that could occur within a complex engineered system. Each accident scenario that is modeled in a PRA is therefore used to represent a category of accident scenarios with similar event sequences, accident progression characteristics, and conditional consequences.
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. 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.

2.C.2. Application of Probabilistic Risk Analysis to Commercial Nuclear Power Plants

Within this scenario-based approach to PRA, risk can be characterized in many ways, depending on the end states of interest for a 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) 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).
Figure 5 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 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 (including LERF) 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 PCA models to quantify conditional measures of the offsite radiological health, environmental, and economic consequences, conditioned on the assumed occurrence of each postulated radiological release category and its representative source term that provides input to the offsite radiological consequence model.4,40
Figure 5. 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) statistical summary measures for consequence metric probability distributions (e.g., mean, median, 95th percentile, and 5th percentile); and (2) risk curves (also termed exceedance frequency curves) that illustrate the
frequency (or probability of frequency if an integrated uncertainty analysis is performed) of exceeding specified consequence levels.\textsuperscript{4,40}

Whether performed as part of an 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).\textsuperscript{ff} Applications of PCA at the USNRC include: (1) regulatory analyses\textsuperscript{2} and backfit\textsuperscript{gg} analyses\textsuperscript{36} 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\textsuperscript{46} or Severe Accident Mitigation Design Alternatives (SAMDA) analyses for new power reactor design stage applications\textsuperscript{47}; and (3) supporting applied research studies—including the “State-Of-the-Art Reactor Consequence Analyses (SOARCA) Project”\textsuperscript{16-18}—which developed much of the technical basis for the state-of-the-art consequence models used in this research.

\textsuperscript{ff} 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).

\textsuperscript{gg} 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.\textsuperscript{39}
2.D. Concurrent Accidents Involving Multiple Operating Reactor Units Co-located at a Shared Site

2.D.1. Key Concepts for Multi-Unit Accident Scenarios

A multi-unit accident scenario is a scenario in which two or more units co-located at a shared site concurrently experience accident scenarios. In principle, these concurrent accident scenarios could be initiated by the same or different events caused by internal or external hazards to which the NPP site is exposed. Moreover, the concurrent accident scenarios in co-located units could occur simultaneously, or they could be staggered in time, with delays between initiating events, pivotal events, and adverse outcomes of interest. Finally, it is possible for concurrent accident scenarios to involve events that are independent across units, or there could be dependent events or interactions between units that link the accident scenarios in time and space.

Returning to Reason’s *Swiss Cheese Model of System Accidents*, a multi-unit accident scenario can be visualized as two or more co-located units concurrently experiencing situations in which the holes of all successive defense layers have aligned, thereby creating an accident trajectory that provides a pathway for initiating events in each unit to result in one or more adverse outcomes of interest. Within this Swiss cheese model, the importance of dependencies or interactions between layers of defense within an individual unit or across multiple co-located units can be better understood. In the context of a single-unit accident scenario, *intra-unit* (within an individual unit) dependencies between one or more successive layers of defense can cause failure of one layer of defense to result in concurrent
failure of multiple layers of defense, thereby making the formation of an accident trajectory within a single unit more likely. Similarly, in the context of a multi-unit accident scenario, \textit{inter-unit} (across or between unit) dependencies between one or more layers of defense can cause failure of one or more layers of defense in one unit to result in concurrent failure of one or more layers of defense in co-located units, thereby making the formation of concurrent accident trajectories in multiple co-located units more likely.

\textbf{2.D.2. 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 multi-unit accident scenarios. 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.\textsuperscript{7}

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 involving SSCs in both units. The conditional consequences of the two-unit radiological release events were estimated by: (1) assuming 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.\(^7\)

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 risk curves; and (2) even for an NPP site with limited sharing of SSCs, two-unit accident scenarios contributed 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%\(^7\).

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 may not be negligible and could be significant, depending on site-specific factors that influence the potential for dependent failure events or adverse interactions across multiple units.

Under existing USNRC regulations, licensees are required to submit a Licensee Event Report (LER) to the USNRC within a specified time after abnormal conditions 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% (391 out of 4207) of LERs reported to the USNRC from 2000 through 2011 affected multiple units at a shared NPP site. This study also included: (1) examples of the types of multi-unit dependencies associated with these events; and (2) 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 be integrated into a single multi-unit PRA model for a shared NPP site. Six categories of inter-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.

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hh Abnormal conditions are those that are beyond the technical specifications that define the conditions under which an NPP is allowed to operate.
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.

2.D.4. The 2011 Fukushima Nuclear Accident: A Salient Example of the Need to Account for 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, which triggered a devastating tsunami that: (1) flooded about a 2,000-kilometer segment of Japan's coast with inundation heights of up to 40 meters; and (2) injured or killed approximately 25,000 people.9

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 Fukushima Daiichi Nuclear Power Station (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 events—a LOOP caused by the earthquake, coupled with subsequent failure of the EDGs caused by tsunami-induced flooding—resulted in a loss of all electrical power to safety-critical SSCs. This category of scenarios is commonly referred to as a station blackout (SBO), which previous NPP PRA studies have demonstrated to be significant contributors to NPP accident risk. In the subsequent 72 hours, the operating reactors melted down, releasing hydrogen and radioactive materials into the surrounding containment structures. 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.

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 one of only 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 international understanding of and communication about the safety significance of events involving various sources of radiation, including commercial 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.

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. Similar to the Richter...
scale for earthquakes, the INES scale is logarithmic; each increase in INES level is thus 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.”9,50 Prior to Fukushima, the Chernobyl nuclear accident was the only nuclear accident to receive an INES rating of Level 7.50 By comparison, the 1979 accident at Three Mile Island—which resulted in severe core damage and partial meltdown of 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.”50

Investigations undertaken in the aftermath of the 2011 Fukushima nuclear accident indicate the accident could have been prevented if both the utility that owned and operated the plant and the nuclear regulatory authority in Japan had followed international best practices and standards with respect to their use of PRA in periodically reevaluating NPP safety considering dynamic external hazards. A number of potential causal factors that underlie this failure to follow evolving international best practices and standards have been identified, including: (1) the nuclear regulatory authority lacked independence from the nuclear industry, as well as other government agencies responsible for promoting the development and use of nuclear power; (2) nuclear officials and professionals were unwilling to take advice from experts outside the nuclear field or to utilize local knowledge about the hazards
to which NPP sites were exposed; and (3) the nuclear safety culture in Japan had become complacent, with many believing that a severe nuclear accident was simply not possible.51

Although there were distinct factors that made the accident more likely to occur in Japan than it may have been in the U.S. or elsewhere, the 2011 Fukushima nuclear accident provides a salient example of the need to consider and account for potential multi-unit accident scenarios. The accident challenged many assumptions that had been used in previous PRA and PCA studies and underscored the importance of many factors that can influence the initiation, progression, or consequences of potential multi-unit accident scenarios. Examples of such influencing factors include: (1) dependencies or interactions across multiple co-located units at a shared NPP site—especially those influenced by large-scale external events—which can cause concurrent failures across multiple units and challenge resources available for mitigating accident scenarios; (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 of released radiological materials; (4) impacts on the affected population of protective actions taken to avert radiological dose and the need to account for psychosocial effects and other non-radiological health effects; and (5) 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.
2.E. Public Health Risks Attributable to Nuclear Accident Scenarios: Evolution of Our State of Knowledge

2.E.1. Radiological Health Effects: Direct Health Effects Due to Ionizing Radiation Exposure

Radiological health effects have traditionally been divided into two categories: (1) deterministic health effects, and (2) stochastic health effects.

2.E.1.I. Deterministic Health Effects

Deterministic health effects were identified soon after Wilhelm Roentgen discovered x-rays in 1895. These health effects occur because of cell death following radiation exposure and include: (1) acute radiation syndrome (ARS), (2) skin burns, (3) hair loss, and (4) death. Since many cells must die to cause sufficient damage to tissues to produce observable effects, high doses of radiation over short periods of time are typically required to exceed dose thresholds for exposed tissues. Deterministic health effects are classified as deterministic because they occur with certainty once threshold doses for exposed tissues are exceeded. Most of these effects occur shortly after threshold doses have been exceeded, with the severity of the effects increasing with increasing dose above the threshold. Since the pattern of signs and symptoms associated with these effects is unique and predictable, the occurrence of deterministic effects can usually be attributed directly to radiation exposure.52
2.E.1.II. Stochastic Health Effects

Stochastic health effects occur because of non-lethal changes in cellular components, especially radiation-induced damage to DNA. Failure of cellular repair mechanisms and subsequent proliferation of affected cells can give rise to two main types of stochastic effects, depending on which type of cell is affected: (1) cancer—which occurs if somatic (body) cells are affected; and (2) hereditary effects—which occur if germ (reproductive) cells are affected and these cells are then passed to future generations. Stochastic health effects were first documented in 1928, after x-rays were shown to induce mutations in germ cells of fruit flies, with the frequency of these effects increasing linearly with increasing radiation dose to the gonads.

Stochastic health effects are classified as stochastic because of inherent randomness in the processes that give rise to them, making it impossible to predict with certainty the individuals who will be affected by them and when they will occur. Whereas the severity of deterministic health effects increases with increasing dose above a specified threshold value, the probability of effect occurrence increases with increasing dose for stochastic health effects. Moreover, it is usually not possible to attribute the occurrence of a stochastic effect in a specific individual directly to radiation exposure for two main reasons: (1) it may take many years for a stochastic effect to appear following radiation exposure; and (2) current methods are unable to distinguish between stochastic effects caused by radiation exposure and those due to other causes.52

However, since radiation exposure can lead to an increased incidence of stochastic effects, it is usually possible to infer an increased risk of stochastic effects in an exposed population using mathematical dose-response models.52 Such models have
been used in both: (1) NPP PRA and PCA studies to estimate the radiological health risk attributable to a spectrum of postulated nuclear accident scenarios; and (2) health studies to estimate the radiological health risks attributable to real-world nuclear accident scenarios based on radiological dose assessments performed after the accidents have occurred.

The dose-response models used for quantifying the lifetime cancer risk attributable to nuclear accident scenarios have been based largely on associations observed in epidemiology studies of well-defined populations exposed to radiation doses that were orders of magnitude greater than what much of the affected public would be exposed to following a nuclear accident. Since high-quality dose-response data for the general population are not available for the range of radiation exposures relevant to nuclear accident scenarios, there is considerable uncertainty about the nature of the true dose-response relationship for these exposures. Extrapolation from the high-dose range of epidemiologic data to make predictions in the low-dose range of interest can thus be performed using a number of plausible alternative mathematical models that provide a reasonable fit to the observed data.

Scientific advisory and governmental bodies continue to endorse the default use of a linear no-threshold (LNT) dose-response model for radiation protection applications—where precaution and conservatism are judged to be prudent. The LNT model is founded on the key hypothesis that any exposure to radiation—no matter how small—incrementally increases lifetime cancer risk proportional to radiation dose. Previous NPP PRA and PCA studies have typically used dose-response models based on the LNT hypothesis to quantify the excess lifetime cancer risk attributable to postulated nuclear accident scenarios. This conservative
approach—if biased—is unlikely to underestimate the radiological risk and thus is more likely to err on the side of protecting the public from radiological health effects.

2.E.1.III. Insights from Probabilistic Analyses and Real-World Nuclear Accidents

In the early stages of commercial NPP development, before society had acquired the substantial base of operational experience we have now, radiological health effects arising from exposure to ionizing radiation were the predominant public health concern related to postulated nuclear accident scenarios. Experience in the aftermath of the atomic bombings at Hiroshima and Nagasaki played a critical role in shaping this concern, and continues to influence radiological risk perceptions among the general public today. Moreover, in the absence of observations about real-world events, early models of nuclear accident scenarios relied on the use of conservative assumptions about: (1) how rapidly such scenarios would progress to core damage and radiological release; (2) the amount of radiological materials that would be released to the environment; and (3) the effectiveness of equipment and human actions at mitigating accident consequences. Thus, early analyses of postulated nuclear accident scenarios estimated that more severe nuclear accident scenarios could result in several thousand early fatalities and injuries among the offsite population if they were to occur. It is important to note that these analyses were performed in the 1950s for reactors that are about half the size of the current fleet of operating reactors.

These estimates were later updated as: (1) more data about the health effects of ionizing radiation exposure became available; (2) understanding of NPP design and
operations improved with the accumulation of operational experience; and (3) probabilistic analysis techniques were developed and advanced. The first NPP PRA and PCA studies were performed in the early 1970s. These studies, which were performed for larger reactors that are still in use today, estimated that early fatalities and injuries for the more severe nuclear accident scenarios would number in the low hundreds for the modeled NPP sites.\(^{19}\) Although these estimates were orders of magnitude lower than previous estimates, radiological health effects remained the predominant public health concern related to postulated nuclear accident scenarios. That the safety goal QHOs address only early fatality risk and latent cancer fatality risk attributable to accidental releases from NPPs reflects this persistent concern.

Experience with real-world nuclear accident scenarios has since challenged the estimates of radiological health risk from previous NPP PRA and PCA studies. Since these earlier studies were performed, three major accidents involving commercial NPPs have occurred: (1) the accident at Three Mile Island in March 1979; (2) the accident at Chernobyl in April 1986; and (3) the accident at Fukushima in March 2011. While there were about 30 early fatalities among emergency response workers that were attributed to acute accidental radiation exposures during the Chernobyl accident,\(^{13}\) no members of the general public have died from acute radiation exposures attributed to these severe nuclear accidents.\(^{14}\) Moreover, apart from detectable increases in the incidence of thyroid cancer cases among young people who were exposed to radiological materials from the Chernobyl accident,\(^{15}\) several health studies indicate that the excess cancer risk attributable to radiation
exposures following these severe nuclear accidents among affected populations is
negligible\textsuperscript{ii} with respect to their background cancer risk.\textsuperscript{14}

These findings are further bolstered by state-of-the-art PCA studies that leveraged
decades of severe accident research and advanced analytical tools to improve our
state of knowledge about severe accident phenomena, accidental radiological
releases, and offsite radiological consequences.\textsuperscript{16-18} As described in Section 2.F., these
contemporary studies indicate that nuclear accident scenarios: (1) progress more
slowly; (2) release less radiological materials to the environment; and (3) result in
fewer adverse radiological health effects than previous studies had indicated.

2.E.2. Non-Radiological Health Effects: Indirect Health Effects Due to
Radiological Materials Released During Nuclear Accident Scenarios

In addition to slower accident progression and release of less radiological materials
to the environment, another reason the radiological health risks attributable to
nuclear accident scenarios are relatively low is that—in both modeled and real-world
accident scenarios—offsite protective actions such as sheltering, evacuation, and
dose-dependent relocation are implemented to avoid or limit accidental radiation
exposures among the affected population. However, implementation of such
protective actions exposes people to a range of other health and safety risks,
including: (1) the risk of injury or death during evacuation; and (2) the risks of
psychosocial effects from permanent or long-term relocation. Implementation of

\textsuperscript{ii} In this context, the term \textit{negligible} means that the increase in cancer risk attributable to accidental
radiation exposures is not large enough to be statistically discernible from normal variation in baseline
cancer rates among the affected population.
offsite protective actions to avert radiological dose thus involves a tradeoff between radiological and non-radiological health risks. Given that a relatively small number of radiological health effects have been attributed to real-world nuclear accidents, it is worth exploring available information about the non-radiological health effects attributed to these accidents to obtain some insights about this risk-risk tradeoff.

A recent study reviewed and summarized the radiological and non-radiological health effects that have been observed in the aftermath of nuclear accidents, with an emphasis on those that have been observed following the Fukushima nuclear accident. A wide range of non-radiological health effects have been reported among populations affected by nuclear accidents, including:

- **Evacuation-related health effects, especially among vulnerable populations forced to evacuate hospitals or nursing care facilities.** About 2,200 inpatients at hospitals and elderly people at nursing care facilities were rapidly evacuated during the first three days of the Fukushima accident. Among these, more than 50 died of complications arising from the evacuation-related impacts on their medical support.

- **Relocation-related health effects, especially among vulnerable populations needing nursing care.** By May 2011, about 170,000 residents had been relocated from the areas surrounding the Fukushima prefecture. The mortality rate among relocated elderly people who needed nursing care increased by a factor of three in the first three months following the accident, and remained about 1-5 times higher than before the accident thereafter. This
increased mortality rate was primarily attributed to the need for repeated relocation and frequent changes in living conditions.

- **Changes in lifestyle and health-related behaviors among relocated populations.** Long-term displacement and living in temporary shelters caused many people who were relocated after the Fukushima nuclear accident to change various aspects of their lifestyle, including: (1) diet, (2) physical exercise, and (3) sleep. Changes in these health-related behaviors have raised concerns about the risk of cardiovascular disease among relocated populations. Analysis of health survey data indicates that relocated individuals are more likely to have: (1) an elevated body-mass index, (2) hypertension, (3) diabetes, and (4) high cholesterol.

- **Psychological health effects among relocated populations and people living in areas with residual radioactive contamination.** Health surveys of people who were relocated following the Fukushima nuclear accident and people living in areas contaminated with residual levels of radioactivity indicate that these groups have an elevated risk of psychological health effects, including: (1) post-traumatic stress disorder (PTSD), (2) other mood and anxiety disorders, and (3) poor health perceptions as measured by subjective health ratings.

- **Psychosocial effects among families and communities.** In addition to psychological health effects at the individual level, complex psychosocial issues have been observed among families and communities affected by the Fukushima nuclear accident. Multiple factors (e.g., displacement, fear of radiation exposure, social stigma) have given rise to three types of discordance that have adversely affected these families and communities: (1) discordance among family members (especially between parents of young children, arising from different perceptions
about the risks of radiation exposure); (2) conflicts between families and communities arising from perceived disparities in governmental restrictions and compensation for damages; and (3) deterioration in relationships between relocated individuals and the residents of communities to which they were relocated, arising from the perceived negative impacts of the sudden population increase in these communities.60

2.E.3. The Need for a Broader Set of Public Health Risk Metrics

The dominant public health consequences arising from the severe nuclear accidents at Three Mile Island, Chernobyl, and Fukushima have not been radiological health effects attributable to direct exposure to radiation released during these accidents. Rather, the dominant public health consequences arising from these accidents have been a broad spectrum of indirect, non-radiological health effects caused in part by implementation of protective actions that aim to avoid or limit radiological dose. Use of these protective actions inherently involves a risk-risk tradeoff between radiological and non-radiological health risks. Yet NPP PRA metrics based on average individual early fatality and latent cancer fatality risk that can be compared against the safety goal QHOs in determining how safe is safe enough are not able to account for these tradeoffs. Instead, NPP PRA models that rely solely on these metrics can take credit for protective actions that reduce radiological health risks, while masking the non-radiological health risks. These observations highlight the need for a broader set of public health risk metrics that can: (1) illuminate risk-risk tradeoffs between radiological and non-radiological health risks; and (2) portray a more complete picture of the public health risks attributable to nuclear accident scenarios.
2.F. The State-of-the-Art Reactor Consequence Analyses (SOARCA) Project

2.F.1. Project Overview

In 2005—six years before the occurrence of the 2011 Fukushima nuclear accident—the USNRC initiated the SOARCA project to: (1) develop integrated state-of-the-art reactor accident progression and offsite radiological consequence models that leveraged both the enhanced state of knowledge about severe accident phenomena and radiological health effects developed over decades of research, as well as modeled improvements in NPP design and operation that had not been reflected in earlier studies; and (2) obtain realistic estimates of the radiological health consequences 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.\textsuperscript{16-18}

This dissertation 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 accident scenarios, and key conclusions for the SOARCA pilot study; follow-on SOARCA studies are also briefly described.

2.F.2. Project Objectives

The overall objective of the SOARCA project was to develop an updated body of knowledge regarding the realistic consequences for important severe nuclear
This overall objective was complemented by a number of supporting objectives, including two that directly pertain to this research16-18:

1. **Incorporate integrated modeling of severe accident progression and offsite consequences using state-of-the-art analytical tools.** The SOARCA project sought to leverage: (1) 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; and (2) advanced capabilities that had been incorporated into state-of-the-art modeling and analytical tools.

2. **Model modifications to NPP design and operation that were not reflected in previous PRA or PCA studies.** The SOARCA project sought to model several enhancements that had been made to NPP design and operation that were not reflected in earlier PRA or PCA studies. Examples of these changes included: (1) system design enhancements; (2) improved training, emergency operating procedures (EOPs), and emergency response plans (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.61

### 2.F.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 Atomic Power Station (hereafter *Peach Bottom*), Unit 2—located about 20 miles south of Lancaster, PA; and (2) Surry Power
Station (hereafter *Surry*), Unit 1—located about 20 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, which is similar to the reactor-containment design used at the FDNPS in Japan. Surry is generally representative of U.S. operating reactors using the Westinghouse pressurized-water reactor (PWR) design with a large, dry containment.

Single-unit accident scenarios were selected for detailed modeling and evaluation using a rigorous process that coupled results and insights from available PRA models for each NPP 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. Single-unit 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 year; or (2) they involved early failure or bypass of containment and their CDF contribution was equal to or greater than 1E-07 per year.¹⁶⁻¹⁸ 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 Table II and Table III in Chapter 1.

To assess the potential benefits of EDMG measures and to provide a basis for comparison to the previous analyses of unmitigated severe accident scenarios, the SOARCA pilot study analyzed each single-unit accident 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 Severe Accident Management Guidelines (SAMGs)—is referred to as the mitigated case. The analysis that does not credit these mitigation measures is referred to as the unmitigated case.\(^{16-18}\) Since a formal human reliability analysis (HRA) was not performed to estimate the failure probabilities for the modeled operator actions in the mitigated cases, only the unmitigated cases are used for the purposes of this dissertation research.

2.F.4. Consideration of Multi-Unit Accident Scenarios

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 multi-unit accident 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 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 (GI)\(^{\text{ii}}\) and referred for further evaluation because it was recognized that

\(^{\text{ii}}\) A generic issue (GI) is an issue involving public health and safety, the common defense and security, or the environment that could affect multiple entities under USNRC jurisdiction. The USNRC documents and tracks resolution of GIs as part of its Generic Issues Program (GIP) that includes three distinct stages: (1) screening, (2) assessment, and (3) implementation. Resolution of GIs may involve: (1) new or revised regulatory requirements; (2) new or revised guidance; or (3) revised interpretation of regulatory requirements or guidance that affect regulated entities.\(^{62}\)
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 USNRC Generic Issues Program. This recommendation was based on: (1) findings from a scoping analysis that aimed to develop a bounding estimate of multi-unit accident 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.63

2.F.5. Key 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 conclusions16-18:

1. When operators are successful in using available onsite equipment during modeled single-unit accident scenarios, they can either: (1) prevent reactor core damage; or (2) delay or reduce radiological releases to the environment.

2. For all modeled single-unit 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 protective actions. If ERPs are implemented as planned and practiced,
offsite protective actions effectively reduce the risk of radiological health consequences attributable to modeled single-unit accident scenarios.

4. For all modeled single-unit 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 single-unit 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 single-unit 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 reactor-containment designs and site characteristics. 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.

2.F.6. SOARCA Uncertainty Analysis

Uncertainty analyses are being performed for specific single-unit 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 offsite radiological 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.\textsuperscript{64}

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.\textsuperscript{64}

\textbf{2.F.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 single-unit accident scenarios involving the Sequoyah Nuclear Plant (hereafter \textit{Sequoyah}). Sequoyah uses a PWR design with an ice condenser containment. Single-unit accident scenarios were thus chosen to specifically challenge this containment type, which is smaller than the large, dry containment used with other PWR NPPs, including Surry.\textsuperscript{65}

The Sequoyah SOARCA study is applying modeling lessons learned and best practices from the SOARCA pilot study.\textsuperscript{26,66} In addition, the effects of using diverse and flexible coping strategies involving portable equipment—which the U.S. nuclear industry has since implemented in response to challenges identified by the 2011 Fukushima nuclear accident\textsuperscript{67}—are also being modeled as part of this study to characterize and evaluate their potential benefits.
2.G. Chapter Summary

This chapter reviewed the available literature germane to this policy analysis that aims to evaluate the effect of a hypothetical expansion in the scope of the safety goal policy to include consideration of: (1) the risk contribution from multi-unit accident scenarios for multi-unit NPP sites; and (2) a broader set of public health risk metrics that includes measures of societal risk for radiological health and non-radiological health consequences. Topics addressed in this chapter included: (1) development, application, and limitations of the USNRC safety goal policy; (2) probabilistic analysis techniques for NPPs, with an emphasis on the traditional scenario-based approach to PRA; (3) insights about multi-unit accident scenarios derived from previous PRAs, operational experience, and the 2011 Fukushima nuclear accident; (4) the evolution of our state of knowledge about the public health risks attributable to nuclear accident scenarios and the need for including a broader set of public health risk metrics in safety goal policy applications; and (5) the objectives, design, and key conclusions from the SOARCA pilot study, which leveraged decades of severe accident research and advances in analytical tools to develop the state-of-the-art consequence models that this research builds upon.

Chapter 3 will describe the models and analytical tools used to perform the policy analysis, including efficient risk estimation models that can be used to calibrate the frequency and conditional consequence results based on SOARCA models to estimate approximately equivalent risk results that can be used to extract meaningful safety goal policy insights, without having to perform a resource-intensive, contemporary full-scope NPP PRA.
Chapter 3. Models and Analytical Tools

3.A. Chapter Introduction and Overview

Chapter 1 introduced central concepts relevant to the U.S. Nuclear Regulatory Commission (USNRC) safety goal policy\(^1\) and probabilistic analysis techniques used to measure attainment of safety goals, as well as limitations in the scope of the safety goal policy that motivate this dissertation research. It further described principal design aspects of a policy analysis that has been performed to evaluate the effects of a hypothetical expansion in the scope of the safety goal policy to address these limitations. Based on a review of the existing literature, Chapter 2 then provided a more detailed description of the background information that is germane to this dissertation research.

This chapter describes the models and analytical tools used to estimate the policy analysis metrics and to address the specific aims for this dissertation research. The chapter explains the need for efficient risk estimation models that can be used to calibrate the frequency and conditional consequence results based on models from the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project\(^{16-18}\) using results from previous full-scope nuclear power plant (NPP) Level 3 probabilistic risk analysis (PRA) studies.\(^{68}\) This calibration is designed to estimate approximately equivalent risk results that can then be used to develop meaningful safety goal policy insights, without having to develop a resource-intensive, contemporary full-scope NPP Level 3 PRA. Analytical tools are summarized, with a detailed treatment of the MELCOR Accident Consequence Code System (MACCS) suite of analytical tools that is used to perform the probabilistic consequence analysis (PCA) element of
a NPP Level 3 PRA. The chapter delineates and illustrates the process and equations used to estimate metrics of interest for the two representative two-unit NPP sites selected as the study population for this dissertation research. Finally, additional consideration is given to how these models—which can be directly applied to NPP sites comprised of two co-located operating reactor units—can be generalized for application to NPP sites comprised of more than two units.
3.B. The Need for Efficient Risk Estimation Models


The USNRC safety goal policy\(^1\) specifies two high-level qualitative safety goals pertaining to commercial NPP operations: (1) an individual risk safety goal specifying in part that individuals should bear no significant additional risk to life and health; and (2) a societal risk safety goal specifying in part that societal risks to life and health should be comparable to or less than risks of generating electricity by viable competing technologies. Each of these qualitative safety goals are supported by a corresponding quantitative health objective (QHO) that can be used to determine whether and to what extent the safety goals have been achieved. The safety goal policy was being developed as the nuclear industry was developing and advancing PRA technology\(^{kk}\) to characterize the public health and safety risks attributable to a spectrum of potential nuclear accident scenarios. A critical issue for this dissertation research is that the QHOs used to measure attainment of the safety goals were developed for comparison with average individual early fatality and latent cancer fatality risk metric results from full-scope\(^{ll}\) NPP Level 3 PRAs that model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential nuclear accident scenarios.

\(^{kk}\) In this context, the term *PRA technology* is used to encompass PRA methods, models, data, and analytical tools.

\(^{ll}\) The safety goal policy identifies two types of accident scenarios that shall be excluded from the scope of such full-scope PRAs used to support safety goal policy applications: (1) accidents initiated by deliberate malevolent acts, including sabotage and terrorist attacks; and (2) accidents involving spent nuclear fuel storage units.
The pilot study for the SOARCA Project\textsuperscript{16-18} used advanced methods, models, data, and analytical tools to develop more realistic estimates of the offsite radiological consequences attributed to potential nuclear accident scenarios for individual reactor units at each of the two NPP sites that comprise the study population for this dissertation research. To leverage decades of severe accident research and advances in analytical tools that improved our state of knowledge about the progression and offsite radiological consequences of potential nuclear accident scenarios, the state-of-the-art source terms and consequence models developed as part of the SOARCA pilot study were selected as the foundation to build upon for this dissertation research. However, an important limitation of this approach is that the SOARCA pilot study was not a full-scope NPP Level 3 PRA; it did not attempt to model a reasonably complete set of accident scenarios intended to represent the full spectrum of potential nuclear accident scenarios. Rather, it is a limited-scope PCA study that performed detailed modeling and integrated analysis of accident progression and offsite radiological consequences for a small set of single-unit accident scenarios that were determined to be important based on PRA model results and expert judgments with respect to: (1) their likelihood of causing core damage—which was assessed using their contribution to core damage frequency (CDF); and (2) their potential for causing significant offsite radiological health consequences due to an early failure or bypass of the containment structure. Since contributions from other potentially risk-significant categories of accident scenarios were deliberately excluded from the SOARCA pilot study by design, direct calculation of risk using the standard method of summing the frequency-weighted consequences across all accident scenario categories would underestimate the total
risk and result in biased inferences. This highlights the need for models that can calibrate the frequency and conditional consequence results for accident scenarios developed as part of both the SOARCA pilot study and this dissertation research to estimate results that are approximately equivalent to those that would be obtained from a full-scope NPP Level 3 PRA, and that can then be used to obtain insights with respect to the safety goal policy.

3.B.2. Introduction to Modeling Approach

Considering the limitations introduced by using models from the SOARCA pilot study for this dissertation research, additional models were needed to perform an adequate evaluation of the effect of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios. An introduction to the approach taken to address each of the areas in need of additional models is provided in this section. More detailed descriptions of the process and equations used to estimate metrics of interest are provided in Sections 3.E, 3.F, and 3.G.

3.B.2.1. Calibrating Results from Models Based on SOARCA Project to Account for Excluded Single-Unit Accident Scenarios

A model was needed to calibrate the single-unit accident scenario frequencies and consequences from the SOARCA pilot study to estimate single-unit accident risk results for each NPP site that would be approximately equivalent to those that would be obtained by performing a full-scope single-unit PRA for each site. This model relied on the key assumption that the limited set of single-unit accident scenarios modeled for each NPP site is representative of the full spectrum of potential single-unit accident scenarios that could occur at each site with respect to
their conditional consequence contribution to selected risk metrics. Under this assumption, single-unit accident scenario frequencies were adjusted using results from previous full-scope NPP PRAs to account for the frequency contribution to selected risk metrics from additional single-unit accident scenarios that have not been modeled, but that are assumed to be represented by the modeled set.

3.B.2.II. Estimating Two-Unit Accident Scenario Frequencies

Since frequencies for only the single-unit accident scenarios were provided as part of the SOARCA pilot study, a model was needed to estimate the frequencies for all modeled two-unit accident scenarios. This model required accounting for four factors: (1) the number of permutations (ordered combinations) for each two-unit accident scenario that could occur; (2) the unconditional frequencies of constituent single-unit accident scenarios that are assumed to occur in the reference unit; (3) the conditional probability of an accident scenario occurring in the co-located unit, given that an accident scenario is assumed to occur in the reference unit; and (4) the conditional probability of a specific accident scenario occurring in the co-located unit, given that a specific accident scenario is assumed to occur in the reference unit.

3.B.2.III. Estimating Two-Unit Accident Scenario Conditional Consequences

Since conditional consequences for only the single-unit accident scenarios were provided as part of the SOARCA pilot study, models were needed to estimate the conditional consequence contribution to selected risk metrics for all selected two-unit accident scenarios under base case and sensitivity analysis assumptions. These models involved implementing a new multi-source modeling capability that was
recently added to the MACCS suite of PCA tools. MACCS models and the general process of using MACCS to perform PCA to estimate conditional consequences are described next in Section 3.C.

3.B.2.IV. Calibrating Results from Models Based on SOARCA Project to Account for Excluded Two-Unit Accident Scenarios

A model was also needed to calibrate the two-unit accident scenario frequencies and consequences from this dissertation research to estimate two-unit accident risk results for each NPP site that would be approximately equivalent to those that would be obtained by performing a full-scope two-unit PRA for each site. This model also relied on the key assumption that the limited set of two-unit accident scenarios modeled for each NPP site is representative of the full spectrum of potential two-unit accident scenarios that could occur at each site with respect to their conditional consequence contribution to selected risk metrics. Under this assumption, two-unit accident scenario frequencies were adjusted using results from previous full-scope NPP PRAs to account for the frequency contribution to selected risk metrics from additional two-unit accident scenarios that have not been modeled, but that are assumed to be represented by the modeled set.

3.B.2.V. Specifying the Figure of Merit

An equation was needed to calculate the relative contribution of multi-unit accident scenarios to the total mean risk for each of the selected risk metrics. This figure of merit (FOM) was selected to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. Of interest to this dissertation research is
whether—for each of the selected risk metrics—the relative contribution of multi-unit accident scenarios to total mean risk is negligible compared to the relative contribution of single-unit accident scenarios.
3.C. Analytical Tools

3.C.1. Summary Description

A summary description of the analytical tools used for each of the major modeling and quantification steps performed as part of this dissertation research is provided below in Table V. More detailed descriptions of the analytical tools used to perform PCA that are not in common use among the general academic community are provided in the subsections that follow.
Table V. Summary Description of Analytical Tools Used for Major Analyses

<table>
<thead>
<tr>
<th>Analytical Tool</th>
<th>Summary Description</th>
<th>Modeling or Quantification Step</th>
<th>Notes</th>
</tr>
</thead>
<tbody>
<tr>
<td>MELMACCS Interface Software Utility Version 2.0.1</td>
<td>MELCOR-to-MACCS interface that extracts source term information needed for PCA from MELCOR output files and generates input file for MACCS.</td>
<td>X</td>
<td>Used to generate reference unit and co-located unit source term input files for MACCS.</td>
</tr>
<tr>
<td>MELCOR Accident Consequence Code System (MACCS) Version 3.10</td>
<td>USNRC-sponsored code for performing PCA of radiological releases from nuclear facilities.</td>
<td>X</td>
<td>Used to perform PCA for all modeled accident scenarios and Aim 3 one-way sensitivity analyses for timing offset parameter used for modeling two-unit accident scenarios.</td>
</tr>
<tr>
<td>Palisade DecisionTools Suite PrecisionTree: Industrial Edition Version 7.5.1</td>
<td>Decision analysis add-in for Microsoft Excel that includes capability for modeling policy alternatives and performing sensitivity analyses.</td>
<td>X</td>
<td>Used to perform Aim 2 one-way sensitivity analyses for level of inter-unit dependence.</td>
</tr>
</tbody>
</table>
3.C.2. MelMACCS Interface Software Utility

The USNRC uses the MELCOR (not an acronym) code for modeling severe accident progression and estimation of source term characteristics. This source term information is contained in MELCOR plot files that cannot be used as a direct input to the offsite radiological consequence model for performing probabilistic accident simulations. The MelMACCS (MELCOR-to-MACCS) interface software utility processes these plot files to extract the source term data needed for performing the probabilistic accident simulations and to generate the corresponding input parameters for the offsite radiological consequence model in MACCS.

For this dissertation research, MelMACCS Version 2.0.1 was used to generate reference unit and co-located unit source term input files for MACCS, using the MELCOR-generated plot files that were developed as part of the SOARCA pilot study.

3.C.3. MELCOR Accident Consequence Code System (MACCS)

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 from nuclear facilities. These factors include: (1) statistical variability in offsite weather conditions over time; (2) atmospheric transport and dispersion (ATD)—including deposition—of released radiological materials; (3) offsite population characteristics; (4) protective actions taken to

\[\text{mm} \text{ The MACCS code includes a simplified model for deposition of atmospheric radioactivity onto water bodies and runoff of deposited material on land into water bodies to support estimation of the contribution to radiological doses from ingestion of contaminated water. However, aqueous releases}\]

100
avert radiological dose; and (5) dose-response models used to estimate numbers of radiological health effects. MACCS was recently enhanced to include a multi-source modeling capability that enables 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. The technical basis that underlies the probabilistic and 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. This section thus focuses on: (1) the new multi-source model that enabled this dissertation research involving the modeling and analysis of multi-unit accident scenarios to be performed; and (2) the general process of using MACCS to perform PCA to estimate conditional consequences for modeled accident scenarios.

### 3.C.3.I. MACCS Multi-Source Model

MACCS was recently enhanced in Version 3.10 to include the capability to model releases from multiple, co-located radiological sources with potentially different accident progression timelines and unique radionuclide inventories. Figure 6 provides an overall conceptual model illustrating the relationships between key inputs and phenomena modeled in MACCS to calculate conditional consequence metrics for modeled two-unit accident scenarios.

directly to water bodies (e.g., runoff, pipe discharges, or discharges to aquifers) are typically not addressed. There are two principal justifications for excluding aqueous releases: (1) the airborne pathway is expected to be dominant for health risks because movement of radionuclides to the accessible environment through aqueous pathways is expected to be slow relative to atmospheric transport; and (2) releases to groundwater or surface water are considered to be easier to interdict.
**Figure 6. Overall Conceptual Model.** A conceptual model illustrates relationships between key inputs and phenomena modeled in MACCS to calculate conditional consequence metrics for modeled 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 VI summarizes these parameters and provides an explanation for how each parameter was treated for this dissertation 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 additional parameters based on the number of source term input files that are specified and the information they contain with respect to: (1) the total number of plume segments; and (2) delay times between plume segments.
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Description</th>
<th>Range of Parameter Values</th>
<th>Units</th>
<th>Treatment</th>
</tr>
</thead>
<tbody>
<tr>
<td>TOTREL</td>
<td>Defines number of plume segments released over all files specified for</td>
<td></td>
<td></td>
<td>Maintained number of plume segments defined in each SOARCA pilot study source term file. MACCS then calculates TOTREL by summing number of plume segments defined in each source term file combined in multi-source model.</td>
</tr>
<tr>
<td></td>
<td>multi-source model. Parameter value is specified when multi-source term files are specified. A constant rate of release is assumed over each plume segment.</td>
<td>2</td>
<td>500</td>
<td>N/A</td>
</tr>
<tr>
<td>MS_LABELS</td>
<td>MelMACCS file name associated with the plume segment.</td>
<td>1-255 characters</td>
<td>N/A</td>
<td>Defined when multi-source term file set specified in MACCS file specifications.</td>
</tr>
<tr>
<td>PLUME_DLY</td>
<td>Start time of plume release relative to MELCOR time frame.</td>
<td>0</td>
<td>2592000</td>
<td>seconds</td>
</tr>
<tr>
<td>NUM_SOURCES</td>
<td>Number of source term files specified. Defined when user specifies multi-source term file set.</td>
<td>2</td>
<td>500</td>
<td>N/A</td>
</tr>
<tr>
<td>SOURCE_TIME</td>
<td>Timing offset for each specified source term file.</td>
<td>0</td>
<td>2592000</td>
<td>seconds</td>
</tr>
<tr>
<td>OFFSET</td>
<td></td>
<td></td>
<td></td>
<td>Timing offset for first source term file was set to 0 to coincide with reference unit accident initiation. Timing offset for second source term file was set to 0 for the base case assumption of simultaneous accidents in the reference unit and co-located unit (Research Aim 1). Timing offset was then varied from 1 to 7 days in one-day increments for sensitivity analyses to evaluate effects of different timing offsets between concurrent accidents in the reference unit and co-located unit (Research Aim 3).</td>
</tr>
</tbody>
</table>
3.C.4. Using MACCS to Perform Probabilistic Consequence Analyses

MACCS is a modular computer code that integrates models embedded within three major modules to sequentially estimate analyst-specified output measures of the conditional consequences attributed to accidental releases of radiological materials from nuclear facilities, including NPP sites. These sequential modules are known as: (1) the ATMOS module; (2) the EARLY module; and (3) the CHRONC module. An overview of each of these modules is provided in the subsections that follow.

3.C.4.I. The ATMOS Module

The ATMOS module integrates probabilistic models of offsite weather conditions with ATD and deposition models to simulate the transport and fate of released radiological materials over time and space. Most source term data needed to characterize the radiological release are automatically entered into the ATMOS module by importing an input file generated by the MelMACCS interface software utility that processes MELCOR files to generate the corresponding ATMOS input parameters. ATMOS input parameters that specify the radiological release include: (1) the total number of chemical groups used to represent the set of modeled radionuclides; (2) the name of each representative chemical group; (3) the set of modeled radionuclides and the chemical group to which each one is assigned; (4) the total number of plume segments that comprise the release, as well as the start time.

\[ This \ section \ represents \ a \ summary-level \ description \ of \ current \ best \ practices \ for \ performing \ PCA \ using \ MACCS.^{26} \]
and duration of each plume segment; and (5) input parameters specified above for the multi-source model.

MACCS models plume dispersion during downwind transport using a segmented, straight-line Gaussian plume model. This model uses a normal distribution for airborne radionuclide concentration in the crosswind and vertical dimensions that are perpendicular to the downwind direction of plume travel. Equations used to implement this Gaussian plume segment ATD model include parameters that depend on several factors, including: (1) wind speed; (2) atmospheric stability class; (3) surface roughness; (4) release height; and (5) the height of the layer in which air mixes with suspended radionuclide particles—which depends upon the temperature profile of the atmosphere. The analyst specifies values for these parameters, along with parameters related to deposition of radionuclides onto land and water bodies under wet (raining) and dry (not raining) conditions, within the ATMOS module.

MACCS includes many options for treatment of offsite weather conditions. Two deterministic options are available: (1) weather is assumed to be constant during and after a radiological release; or (2) weather is defined by an analyst-specified set of 120 weather data points. However, MACCS analyses in support of NPP Level 3 PRA or PCA studies utilize one of its probabilistic options that involve sampling from weather bins that are defined using site-specific weather data. Each weather bin includes a set of meteorological observations that are similar with respect to three factors: (1) wind speed; (2) atmospheric stability class; and (3) precipitation. When one of these sampling options is implemented, a probabilistic accident simulation using MACCS represents a collection of simulations—one for each
sample from an analyst-specified weather bin. For this dissertation research, the probabilistic option used in the SOARCA pilot study was implemented. In this option, 36 discrete weather bins were defined based on hourly meteorological data that had been collected over the course of one year (8,760 hours) from stations at each NPP site. The SOARCA pilot study implemented a non-uniform sampling strategy that allows the analyst to specify a different number of random samples for each weather bin that accounts for the relative frequency with which each weather bin has been observed to occur at the modeled NPP site. In total, this approach resulted in about 1,000 weather trials per probabilistic accident simulation for each combination of NPP site and accident scenario (994 for Peach Bottom and 1020 for Surry).

Within the ATMOS module, the analyst also defines the spatial grid used to model the region of interest surrounding the modeled NPP site by specifying: (1) the number of compass sectors used to divide the 360-degree polar coordinate system; (2) the number of radial intervals extending from the point of release to the outer boundary of the region of interest; and (3) the distance from the point of release to the outer boundary of each radial interval. Specifying a larger number of elements to define the spatial grid increases the resolution of an analysis, but also increases its

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These probabilistic sampling techniques are used to account for uncertainty about: (1) when an accidental release will occur within a given time-period (typically one year of reactor operation for NPP PRA and PCA studies); and (2) what the prevailing weather conditions will be for the duration of the release(s). Using these probabilistic sampling techniques, MACCS can generate probability distributions for selected consequence metrics that reflect variability in consequences arising from statistical variability in weather conditions over time. The mean conditional PCA results then represent the weighted average for the selected consequence metric over all modeled weather conditions that are assumed to be possible when an accidental release occurs, weighted by the probability of each individual weather trial.
computational demands. The spatial grid defined in the ATMOS module also applies to analyses that implement the EARLY and CHRONC modules described below.

Other than the additional multi-source model parameters identified in Table VI for two-unit accident scenarios, this dissertation research used the same values for all ATMOS input parameters as those used in the SOARCA pilot study for all modeled accident scenarios. The technical basis for selected input parameter values is documented in accessible resources and has been subjected to peer review by a panel of individuals with relevant expertise.\textsuperscript{16-18}

3.C.4.II. The EARLY Module

The EARLY module integrates models of offsite population distribution, protective actions, human exposure and dosimetry, and dose-response to estimate multiple measures of conditional consequences arising from radiation exposures that occur during the early phase of response to an accidental release of radiological materials.\textsuperscript{pp} MACCS enables the analyst to define multiple emergency response cohorts, with each cohort representing a subset of the offsite population that is expected to exhibit similar protective action behaviors in response to guidance or direction from offsite response organizations (OROs). In this context, protective action behaviors are represented by several parameters, including: (1) radiation shielding and exposure factors for each modeled activity and exposure pathway; (2) delays in implementing emergency response protective actions; and (3) evacuation

\textsuperscript{pp} Per U.S. Environmental Protection Agency (USEPA) guidance, the \textit{early phase} is defined as the beginning of a radiological incident when immediate decisions for effective use of protective actions are required and must therefore be based primarily on the status of the radiological incident and the prognosis for worsening conditions. This phase may last from hours to days.
speed and direction for populations that are assumed to evacuate. Activities modeled in the EARLY module include normal activity (which assumes no protective actions are taken) and four protective actions: (1) evacuation; (2) sheltering; (3) dose-dependent relocation; and (4) ingestion of potassium iodide (KI) as a supplementary protective action to reduce the radiological dose to the thyroid gland by blocking the uptake of radioactive isotopes of iodine. Five exposure pathways are modeled: (1) direct exposure to external radiation from the plume of released radiological materials (termed cloudshine); (2) inhalation of radioactivity in the plume; (3) direct exposure to external radiation from ground contamination (termed groundshine); (4) inhalation of resuspended radioactivity; and (5) contamination of skin and clothing. Exposure durations are typically based on site-specific evacuation time estimate (ETE) analyses and early phase protective action models. MACCS produces conditional consequence results for each individual emergency response cohort, as well as overall results for the offsite population that represent a weighted average of the individual cohort results, weighted by the fraction of the offsite population assigned to each cohort. Overall results are used for safety goal policy applications and were used to address the specific aims for this dissertation research.

For the SOARCA pilot study and this dissertation research, six emergency response cohorts were defined for each NPP site:

- **Cohort 1 – General Public:** The general public residing within the 0-10 mile emergency planning zone (EPZ).
• **Cohort 2 – Shadow Evacuation**: People residing within the 10-20 mile radial area beyond the 0-10 mile EPZ that are assumed to evacuate, even when not directed to do so by the OROs.

• **Cohort 3 – Schools**: School-age populations attending elementary schools, middle schools, and high schools located within the 0-10 mile EPZ.

• **Cohort 4 – Special Facilities**: Populations residing in special facilities (e.g., healthcare facilities, assisted living facilities, and correctional facilities) located within the 0-10 mile EPZ.

• **Cohort 5 – Evacuation Tail**: The general public residing within the 0-10 mile EPZ that comprise the last 10% of this population to evacuate the EPZ.

• **Cohort 6 – Non-Evacuating Public**: The general public residing within the 0-10 mile EPZ that are assumed to not evacuate when the OROs advise or direct them to do so, and instead maintain normal activity.

Within the EARLY module, the analyst also specifies radiological health effects and corresponding target organs of interest. For each health effect and target organ, the analyst specifies parameters for a dose-response model used to estimate the numbers of health effect cases arising from doses accumulated during the early phase of response to an accidental radiological release. Whereas threshold-based dose-response models are used for deterministic health effects (e.g., early fatalities), the analyst can specify whether a threshold-based model or a dose-response model based on the linear no-threshold (LNT) hypothesis is used for stochastic health effects (e.g., latent cancer fatalities). In the SOARCA pilot study, sensitivity analyses were performed to evaluate the impact of using plausible alternative dose-response models for stochastic health effects. 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. However, for this dissertation 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.

With this exception, this dissertation research used the same values for all EARLY input parameters as those used in the SOARCA pilot study for all modeled accident scenarios. The technical basis for selected input parameter values is documented in accessible resources and has been subjected to peer review by a panel of individuals with relevant expertise.16-18

3.C.4.III. The CHRONC Module

The CHRONC module integrates models of offsite population distribution, protective actions, human exposure and dosimetry, dose-response, and regional economies to estimate analyst-specified measures of conditional consequences arising from contamination of commodities and radiation exposures that occur during the
intermediate\(^{\text{99}}\) and late (recovery)\(^{\text{10}}\) phases of response to an accidental release of radiological materials.

Intermediate phase protective actions modeled in the CHRONC module include: dose-dependent relocation or interdiction and dose-dependent bans on agricultural products. Late phase protective actions or recovery actions modeled in the CHRONC module include multiple dose-dependent actions, including: (1) population relocation; (2) temporary land interdiction; (3) permanent land interdiction (termed condemnation\(^{\text{11}}\)); (4) decontamination of land; and (5) bans on agricultural products or condemnation of farmland. The CHRONC module models three exposure pathways: (1) groundshine; (2) inhalation of resuspended radioactivity; and (3) ingestion of contaminated food and water. A lifetime exposure duration of 50 years is typically assumed for the recovery phase and was assumed for this dissertation research.

CHRONC models of protective action decisions are based on two sets of independent decisions: (1) habitability decisions—decisions regarding whether land at a specific time and location is suitable for people to return to for living; or (2) farmability decisions—decisions regarding production of agricultural commodities. The habitability dose criterion is defined by a maximum dose and an exposure period to

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\(^{\text{99}}\) Per USEPA guidance, the intermediate phase is defined as the period beginning after the radiological source and releases have been brought under control and reliable environmental measurements are available for use as a basis for decisions on protective actions and extending until these additional protective actions are no longer needed. In principle, this phase may overlap with the early phase and late phase and may last from weeks to months. However, MACCS imposes constraints that prohibit the overlap of phases in consequence models and thus require the analyst to define phases such that each subsequent phase begins at the end of the preceding phase. For the SOARCA pilot study and this dissertation research, an intermediate phase of zero duration was assumed, such that only the early and late (recovery) phases were modeled.

\(^{\text{10}}\) Per USEPA guidance, the late (recovery) phase is defined as the period beginning when recovery actions designed to reduce radiation levels in the environment to acceptable levels are commenced and ending when all recovery actions have been completed. This phase may extend from months to years.
receive that dose. This criterion is an analyst-specified input parameter in the
CHRONC module that is used to model the return of relocated individuals to land
areas contaminated with various levels of radioactivity; the value of this input
parameter typically corresponds with the USEPA protective action guide (PAG) for
relocation, unless a state or local authority has established a more restrictive
criterion.

Habitability decisions can result in four possible outcomes: (1) land is immediately
habitable; (2) land is habitable after decontamination; (3) land is habitable after
decontamination and interdiction; or (4) land is not deemed habitable after 30 years
of interdiction, and is therefore considered condemned. These decisions are made
using a decision tree approach. The first decision is whether land is immediately
habitable. If it is, no further actions are needed, and relocated individuals return to
their homes and receive a dose from any residual radioactive contamination for the
entire duration of the recovery phase. If land is not habitable, the first option
considered is to decontaminate to the lowest level of dose reduction, which is also the
least expensive to implement. If this level is sufficient to restore the land to
habitability, then this level of decontamination is assumed to occur. After
decontamination, relocated individuals return to their homes and receive a dose
from any residual radioactive contamination for the remainder of the recovery
phase. If the first level of decontamination is insufficient to restore habitability, then
successively higher levels are considered, up to the maximum number of three levels
that can be specified in MACCS. If the highest level of decontamination is
insufficient, then interdiction for up to 30 years is assumed following
decontamination. During the interdiction period, radioactive decay and weathering
processes naturally reduce the residual radioactivity and dose rates that individuals would receive upon returning to the area. If the highest level of decontamination followed by interdiction is sufficient to restore habitability, then it is implemented and relocated individuals can return and receive a dose from any residual radioactive contamination for the remainder of the recovery phase. If habitability cannot be restored by any of these actions, then the land is considered condemned.

Farmability decisions regarding whether land is suitable for production of agricultural commodities first require a determination that the land is habitable. In other words, land cannot be used for farming if it is not suitable for people to occupy. In addition, farmland must be able to grow crops or produce dairy products that meet U.S. Food and Drug Administration (FDA) requirements. If farmland is determined to be both habitable and farmable, a food chain model is used to determine doses that result from ingesting commodities produced on this land. However, since experts believe the availability of agricultural commodities in the U.S. is sufficient to preclude the need for individuals to consume contaminated products, MACCS now allows the analyst to bypass this food chain/ingestion model altogether; based on these expert judgments, this bypass feature was utilized in both the SOARCA pilot study and in this dissertation research.

This dissertation research used the same values for all CHRONC input parameters as those used in the SOARCA pilot study for all modeled accident scenarios. The technical basis for selected input parameter values is documented in accessible resources and has been subjected to peer review by a panel of individuals with relevant expertise.\textsuperscript{16-18}
3.C.4.IV. Two-Unit Probabilistic Accident Simulations

For this dissertation research, MACCS Version 3.10 was used to perform eight discrete probabilistic accident simulations for each of 25 two-unit accident scenarios (nine for Peach Bottom and 16 for Surry) to calculate the conditional consequence contribution to selected risk metrics. One simulation represented the base case analysis that assumed the constituent accident scenarios that comprise each two-unit accident scenario occur simultaneously, with the value of the SOURCE TIME OFFSET parameter in Table VI set to 0 seconds. The remaining seven simulations represented sensitivity cases for two-unit accident scenarios in which the timing offset (delay time) between concurrent accident scenarios involving both units was varied from 1 to 7 days, in one-day increments. Collectively, this resulted in a total of 200 two-unit accident simulations across both NPP sites.
3.D. Methods Overview

Figures 7, 8, and 9 respectively illustrate the processes and information used to implement models developed for: (1) estimating the contributions to selected risk metrics from single-unit accident scenarios; (2) estimating the contributions to selected risk metrics from two-unit accident scenarios; and (3) estimating selected risk metrics for each policy alternative evaluated in this dissertation research and calculating the FOM used to evaluate the effect of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. In addition to the FOM calculation, Figure 9 also illustrates the process and information used to estimate the margin to each safety goal QHO (the early fatality risk QHO and the latent cancer fatality risk QHO) for each policy alternative. This information is used to provide supplementary insights about the effect of this expansion in the scope of the safety goal policy that are specific to the individual radiological health risk perspective considering the existing QHOs; these supplementary insights are used to augment those derived from the selected FOM.

More detailed descriptions about the process, equations, variables, and data sources that correspond to each figure are provided in the following subsections. Although these figures and descriptions focus on the application of these methods to a two-unit case study involving two representative NPP sites, generalization of these methods for application to NPP sites comprised of more than two units is considered in Section 3.H.
Figure 7. Process for Estimating the Contribution from Single-Unit Accident Scenarios to Selected Risk Metrics. Steps 1 through 5 are used to estimate the contribution from single-unit accident scenarios to selected risk metrics.
Figure 8. Process for Estimating the Contribution from Two-Unit Accident Scenarios to Selected Risk Metrics. Steps 6 through 10 are used to estimate the contribution from two-unit accident scenarios to selected risk metrics.
Figure 9. Process for Estimating Selected Risk Metrics for Each Policy Alternative and for Calculating the Figure of Merit.

Steps 11 and 13 are respectively used to estimate values for selected risk metrics under each policy alternative and to calculate the figure of merit. Step 12 is used to estimate the margin to each safety goal QHO under each policy alternative.
3.E. Estimation of the Contribution from Single-Unit Accident Scenarios to Selected Risk Metrics (Figure 7)

3.E.1. Select and Model Important Single-Unit Accident Scenarios (Step 1)

The first step toward estimating the contribution from single-unit accident scenarios to selected risk metrics is to select and model the single-unit accident scenarios that are judged to be important contributors to risk. For this dissertation research, this step was performed as part of the SOARCA pilot study.\textsuperscript{16-18}

In that study, single-unit accident scenarios were selected for detailed modeling and evaluation using a rigorous process that coupled results and insights from available PRA models for each NPP 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.

Single-unit 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 year; or (2) they involved early failure or bypass of containment \textit{and} their CDF contribution was equal to or greater than $1E^{-07}$ per year.\textsuperscript{16-18} Using this approach, seven single-unit accident scenarios were selected for detailed modeling and evaluation under the SOARCA pilot study—three for Peach Bottom and four for Surry. These accident scenarios are summarized in Table II and Table III in Chapter 1.
3.E.2. Estimate Single-Unit Accident Scenario Frequencies and Conditional Consequences (Step 2)

3.E.2.I. Estimate Single-Unit Accident Scenario Frequencies (Step 2a)

To estimate the expected annual frequency of each single-unit accident scenario \( i \) selected for detailed modeling and evaluation \( (F_i) \), the SOARCA pilot study utilized information in plant-specific PRA models to calculate the contribution to CDF from similar categories of accident scenarios represented by the selected single-unit accident scenarios. More detailed descriptions about the methods used to estimate the single-unit accident scenario frequencies are provided in the SOARCA pilot study documentation.\(^{16-18}\) The mean frequency for each selected single-unit accident scenario is provided in Table II in Chapter 1.

3.E.2.II. Estimate Single-Unit Accident Scenario Conditional Consequences (Step 2b)

To estimate the expected conditional consequences for each selected single-unit accident scenario \( i \) \( (C_i|i) \), the SOARCA pilot study developed detailed and realistic models of accident progression using the MELCOR code to estimate scenario-specific source term information. This information was used as an input to detailed and realistic offsite radiological consequence models that were developed for each single-unit accident scenario using the MACCS suite of analytical tools. Using MACCS, the SOARCA pilot study performed probabilistic accident simulations to estimate the
mean conditional consequences attributable to each single-unit accident scenario, conditioned on the assumed occurrence of the scenario. More detailed descriptions about the methods used to estimate the single-unit accident scenario conditional consequences are provided in the SOARCA pilot study documentation, along with the technical basis for key modeling and input parameter value choices.16-18

These conditional consequence calculations were originally performed outside the boundary for this thesis research. However, since the SOARCA pilot study calculated a different set of conditional consequence metrics, the probabilistic accident simulations using MACCS had to be performed again for each modeled single-unit accident scenario to generate estimates of the conditional consequence contribution to the broader set of public health risk metrics that were selected for this dissertation research. For these simulations, the only change made to the offsite radiological consequence models in MACCS was to ensure the complete set of conditional consequence metrics needed for this dissertation research were calculated and reported in the MACCS output files.

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These accident simulations used probabilistic sampling techniques to account for what has been shown in previous studies to be dominant contributors to uncertainty in the offsite public health consequences for a given accidental release: (1) when the accident will occur within a specified time-period (typically one year of reactor operation for NPP PRA and PCA studies); and (2) what the prevailing weather conditions will be for the duration of the release(s). Using these probabilistic sampling techniques, MACCS can generate distributions for selected consequence metrics that reflect variability in conditional consequences arising from statistical variability in weather conditions over time. The mean conditional consequence results then represent the probability-weighted average for the selected consequence metric over all modeled weather conditions that are assumed to be possible when an accident occurs.
3.E.3. Estimate Unadjusted Single-Unit Accident Scenario Risk (Step 3)

Consistent with well-established procedures for performing NPP PRAs, the unadjusted contribution to each selected risk metric from each single-unit accident scenario $i$ is represented by its frequency-weighted conditional consequences. Unadjusted single-unit accident scenario risk is therefore calculated using Equation:

$$ (R^u_{i}) = F^s_i \cdot (C^s_i | i) $$

- $(R^u_{i}) = \text{unadjusted mean risk contribution from single-unit accident scenario } i \text{ to each selected risk metric. In this context, the term } \textit{unadjusted} \text{ means the risk contribution has } \textbf{not} \text{ been calibrated using results from previous full-scope NPP Level 3 PRAs to account for the frequency contribution from other single-unit accident scenarios that are not modeled, but that are assumed to result in similar conditional consequence distributions as scenario } i, \text{ and therefore are assumed to be represented by scenario } i.$

- $F^s_i = \text{mean frequency for single-unit accident scenario } i.$ Estimates for the Peach Bottom and Surry single-unit accident scenario mean CDF contributions were developed as part of the SOARCA pilot study\textsuperscript{16-18} and are provided in Table II in Chapter 1.

- $(C^s_i | i) = \text{mean conditional consequence, conditioned on the assumed occurrence of representative single-unit accident scenario } i.$ This represents the conditional consequence contribution to each selected risk metric from the category of single-unit accident scenarios that scenario $i$ is assumed to represent.
3.E.4. Estimate Single-Unit Frequency Adjustment Factor and Adjusted Single-Unit Accident Scenario Risk (Step 4)

3.E.4.I. Estimate Single-Unit Frequency Adjustment Factor (Step 4a)

An adjustment factor is needed to calibrate 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 representative category that have not been modeled and analyzed. This approach assumes that each single-unit accident scenario \( i \) results in conditional consequence distributions that are like 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 relative contribution to total single-unit accident frequency of single-unit accident scenarios that are not modeled is the same for all categories of single-unit accident scenarios that are represented. The single-unit frequency adjustment factor can be estimated using Equation (2):

\[
\alpha_s = \frac{F_{\text{total}}}{\sum F_i} \tag{2}
\]

- \( \alpha_s \) = global single-unit frequency adjustment factor.
- \( F_{\text{total}} \) = mean total single-unit accident frequency from all single-unit accident scenarios initiated by internal and external hazards. Estimates of the total single-unit CDFs from accident scenarios initiated by internal events, fires, and seismic events for Peach Bottom and Surry were provided in the NUREG-1150 study\textsuperscript{21} that documented full-scope NPP Level 3 PRAs for five representative 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 sufficient. Moreover, since the FOM represents a relative measure of the effect of including the risk contribution from multi-unit accident scenarios, it can be mathematically demonstrated that this parameter does not influence the FOM, because it appears in both the numerator and denominator of the ratio measure, and is therefore eliminated. This parameter therefore only influences absolute measures, which for this dissertation research include: (1) estimates of selected risk metrics for each policy alternative; and (2) estimates of the QHO margins for each policy alternative.

- $\sum_i F_i^s$ = 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.

3.E.4.II. Estimate Adjusted Single-Unit Accident Scenario Risk (Step 4b)

Once the single-unit frequency adjustment factor has been estimated, the adjusted contribution to each selected risk metric from each single-unit accident scenario $i$ is estimated using Equation (3):

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

- $R_i^s$ = adjusted mean risk contribution to each selected risk metric from the category of single-unit accident scenarios that scenario $i$ is assumed to represent. In this context, the term *adjusted* means the risk contribution has been calibrated using results from previous full-scope NPP Level 3 PRAs to account for the frequency
contribution from other single-unit accident scenarios that are not modeled, but that are assumed to result in similar conditional consequence distributions as scenario \( i \), and therefore are assumed to be represented by scenario \( i \).

### 3.E.5. Estimate Total Single-Unit Accident Risk (Step 5)

At the level of individual reactor units, the modeled single-unit accident scenarios are defined to be **mutually exclusive**. This means that only one of the modeled scenarios can occur at a time in a unit. Under this mutually exclusive condition, the total contribution to each selected risk metric from all single-unit accident scenarios is estimated using Equation (4):

\[
R_{total}^s = \sum_i R_i^s \tag{4}
\]

- \( R_{total}^s \) = total mean risk contribution to each selected risk metric from all single-unit accident scenarios.
3.F.  Estimation of the Contribution from Two-Unit Accident Scenarios to Selected Risk Metrics (Figure 8)

3.F.1. Select and Model Important Two-Unit Accident Scenarios (Step 6)

The first step toward estimating the contribution from two-unit accident scenarios to selected risk metrics is to select and model the two-unit accident scenarios that are judged to be important contributors to risk. The process used to select the two-unit accident scenarios for modeling and analysis as part of this dissertation research was described in Section 1.E.3. in Chapter 1. Using the multi-source model in MACCS, 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 for each NPP site in the SOARCA pilot study, assuming the two units at each site are identical and subject to the same set of single-unit accident scenarios. A total of 25 two-unit accident scenarios (nine for Peach Bottom and 16 for Surry) were constructed from all possible combinations of single-unit accident scenarios that were modeled for each NPP site as part of the SOARCA pilot study.

3.F.2. Estimate Two-Unit Accident Scenario Frequencies and Conditional Consequences (Step 7)

3.F.2.I. Estimate Two-Unit Accident Scenario Frequencies (Step 7a)

To estimate the expected annual frequency of each two-unit accident scenario $ij$, four parameters need to be accounted for: (1) the number of permutations (ordered combinations) for each two-unit accident scenario that could occur, which is two for each two-unit NPP site; (2) the unconditional frequency of single-unit accident
scenario \( i \) occurring in the reference unit; (3) the conditional probability of a concurrent accident scenario occurring in the co-located unit, given that accident scenario \( i \) is assumed to occur in the reference unit; and (4) the conditional probability of accident scenario \( j \) concurrently occurring in the co-located unit, given that single-unit accident scenario \( i \) is assumed to occur in the reference unit and given that a concurrent accident scenario is assumed to occur in the co-located unit.

Two alternative approaches to estimating the last parameter were considered. The first approach was to assign an equal conditional probability to each of the single-unit accident scenarios that could occur in the co-located unit, given that a specific single-unit accident scenario is assumed to occur in the reference unit. 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 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 Interfacing Systems Loss-Of-Coolant Accident (ISLOCA) scenario—which involves a bypass of the containment structure and has the lowest unconditional frequency of all modeled single-unit accident scenarios at 3E-08 per year—should still be less likely to occur in the co-located unit than a Long-Term Station Blackout (LTSBO) scenario—which is much more likely with an unconditional frequency of 2E-05 per year—regardless of which single-unit accident scenario is assumed to occur in the reference unit. Based on the adopted approach, the frequency of each two-unit accident scenario is estimated using Equation (5):
\[ F_{ij}^t = 2 \cdot F_i^s \cdot \beta_i \cdot \left( \frac{F_j^s}{\sum_i F_i^s} \right) \]  \hspace{1cm} (5)

- \( F_{ij}^t \) = mean frequency for 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.

- The constant (2) represents the number of permutations for each two-unit accident scenario that could occur, which is two for each two-unit NPP site. Figures 2 and 3 in Chapter 1 respectively illustrate: (1) nine two-unit accident scenario models for Peach Bottom Unit 2 and Unit 3 that can be created by combining the three single-unit accident scenarios evaluated for Peach Bottom as part of the SOARCA pilot study, assuming the Unit 2 accident scenario occurs first; and (2) 16 two-unit accident scenario models for Surry Unit 1 and Unit 2 that can be created by combining the four single-unit accident scenarios evaluated for Surry, assuming the Unit 1 accident scenario occurs first. Since combinations of two-unit accident scenarios in which an accident scenario in the second unit for each NPP site (Unit 3 for Peach Bottom or Unit 2 for Surry) occurs first are also possible, the total number of two-unit accident scenarios for each NPP site is twice the number illustrated (18 for Peach Bottom and 32 for Surry). In principle, the order in which the accident scenarios involving two units occur could matter, especially if there are important differences between the two units. Therefore, it would be more appropriate to consider the permutations (ordered combinations) of two-unit accident scenarios that could occur. However, since the two units at each NPP site are assumed to be identical, these additional two-unit accident scenarios that are not illustrated in
Figures 2 and 3 would result in identical conditional consequences. This assumption of identical units—which is reasonable for both NPP sites—thus eliminates the need to develop unique consequence models for these additional two-unit accident scenarios and reduces the total number of two-unit models by a factor of two. However, while the assumption of identical units reduces the total number of two-unit accident scenario models needed for each NPP site by a factor of two, this does not eliminate the need to account for the contribution of the other half of the two-unit accident scenarios to the frequency of two-unit accident scenarios; the constant (2) accounts for this contribution.

- $\beta_i$ = conditional probability of an accident occurring in the co-located unit, given that single-unit accident scenario $i$ is assumed to occur in the reference unit. While it is expected that two-unit accident scenarios comprised of different reference unit accident scenarios will have different values of $\beta_i$, information about the values of such conditional probabilities is not available in the existing literature. For this dissertation research, a global average conditional probability ($\beta$) is thus assumed to apply across all multi-unit accident scenarios. 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 scenario, given that a core damage accident scenario is assumed to occur in the reference unit. Based on results and insights from previous multi-unit NPP PRA studies $^7$ and operational experience data $^{12}$ 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.
• \( \left( \frac{F_j}{\sum_i F_i} \right) \) = conditional probability of single-unit accident scenario \( j \) concurrently occurring in the co-located unit, given that: (1) single-unit accident scenario \( i \) is assumed to occur in the reference unit; and (2) a concurrent accident scenario is assumed to occur in the co-located unit. This parameter 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, and reflects the unconditional relative likelihood of each single-unit accident scenario.

3.F.2.II. Estimate Two-Unit Accident Scenario Conditional Consequences (Step 7b)

To estimate the expected conditional consequences for each selected two-unit accident scenario \( ij \) \( (C_{ij}^t \mid ij) \), the multi-source model in MACCS described in Section 3.C.3 was implemented to combine the corresponding source term input files for accident scenario \( i \) and accident scenario \( j \). Using MACCS, probabilistic accident simulations were then performed to estimate the mean conditional consequences attributable to each two-unit accident scenario, conditioned on the assumed occurrence of the scenario.

For this dissertation research, the only multi-source model parameter in Table VI 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 scenario in the co-located unit needed to be defined to fully specify the multi-source model for all two-unit accident scenarios. There were two reasons for this: (1) this dissertation 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 fixed at two, since accident scenarios involving spent fuel pool units and dry cask storage facilities are excluded from this dissertation research; the NUM_SOURCES parameter was thus always set to 2 for all two-unit accident scenario simulations.

To evaluate the effect of variation in the assumed SOURCE TIME OFFSET parameter on results for selected conditional consequence metrics, the following analyses were performed with alternative values assigned to the SOURCE TIME OFFSET parameter for the co-located unit:

- **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 the co-located unit.

- **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. A limited set of trial sensitivity analyses were also performed to evaluate the effect of varying the
timing offset parameter from 1 hour to 24 hours in one-hour increments. Only results for one-way sensitivity analyses using the 1-day to 7-day range in one-day increments are presented for three reasons: (1) the 7-day range was judged to be more appropriate for the expected termination of major releases for multi-unit accident scenarios; (2) conditional consequence 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 (3) one-day increments appeared to provide sufficient resolution to characterize patterns or trends.

3.F.3. Step 8: Estimate Unadjusted Two-Unit Accident Scenario Risk

As with the unadjusted single-unit accident scenario risk, the unadjusted contribution to each selected risk metric from each two-unit accident scenario \( ij \) is represented by its frequency-weighted conditional consequences. Unadjusted two-unit accident scenario risk is therefore calculated using Equation (6):

\[ \text{Risk}_{ij} = \sum_{k} \left( \frac{f_{k}}{\text{Consequence}_{k}} \right) \]

An initial upper limit of 24 hours was initially selected to coincide with the standard mission time used in NPP PRA models. The mission time is the time period that a system or component is required to operate to successfully perform its intended function. The mission time influences the criteria used to define successful operation of systems or components, and therefore influences estimates for failure probability parameters that are used in calculating accident scenario frequencies. Although the use of a 24-hour mission time continues to be standard practice in NPP PRAs, there is ongoing debate about whether this should be extended to 72 hours or longer, based on lessons learned from the 2011 Fukushima nuclear accident.

Although it is common practice to establish a convergence criterion when performing Monte Carlo simulations to determine the number of trials needed for results of interest to converge or stabilize, no specific convergence criterion was established for determining whether the 7-day interval was sufficient for conditional consequence results to converge or stabilize. Rather, inspection of the results was used to determine whether sufficient convergence had been observed as the timing offset approached the upper limit of the interval. Using this approach, it was judged that the rate of change in all conditional consequence results as the timing offset approached 7 days was not significant enough for the purposes of this research to warrant extending the interval beyond 7 days.
\[
(R_{ij}^t)_u = F_{ij}^t \cdot (C_{ij}^t|ij)
\]  \hspace{1cm} (6)

- \((R_{ij}^t)_u\) = unadjusted mean risk contribution from two-unit accident scenario \(ij\) to each selected risk metric. In this context, the term \textit{unadjusted} means the risk contribution has \textbf{not} been calibrated using results from previous full-scope NPP Level 3 PRAs to account for the frequency contribution from other two-unit accident scenarios that are not modeled, but that are assumed to result in similar conditional consequence distributions as scenario \(ij\), and therefore are assumed to be represented by scenario \(ij\).

- \(F_{ij}^t\) = mean frequency for two-unit accident scenario \(ij\).

- \((C_{ij}^t|ij)\) = mean conditional consequence, conditioned on the assumed occurrence of representative two-unit accident scenario \(ij\). This represents the conditional consequence contribution to each selected risk metric from the category of two-unit accident scenarios that two-unit accident scenario \(ij\) is assumed to represent.

3.F.4. \textit{Step 9: Estimate Two-Unit Frequency Adjustment Factor and Adjusted Two-Unit Accident Scenario Risk}

3.F.4.I. \textit{Estimate Two-Unit Frequency Adjustment Factor (Step 9a)}

Like the single-unit accident scenario case, an adjustment factor is needed to calibrate the two-unit accident scenario risk estimates to account for the contribution to the frequency element of the risk triplet from other two-unit accident scenarios in each representative category that have not been modeled and analyzed. This approach assumes that each two-unit accident scenario \(ij\) results in conditional consequence distributions that are like those that would result from each of the
other two-unit accident scenarios that 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 relative contribution to total two-unit accident frequency of two-unit accident scenarios that are not modeled is the same for all categories of two-unit accident scenarios that are represented.

The two-unit frequency adjustment factor can thus be estimated using Equation (7):

\[
\alpha^t = \left( \frac{2 \cdot \beta \cdot F_{t \text{total}}}{\sum_{ij} F_{ij}^t} \right)
\]

(7)

- \( \alpha^t \) = global two-unit frequency adjustment factor.
- \( 2 \cdot \beta \cdot F_{t \text{total}} \) = mean total two-unit accident frequency from all two-unit accident scenarios initiated by internal and external hazards. Estimates of the total single-unit CDFs from accident scenarios initiated by internal events, fires, and seismic events for Peach Bottom and Surry were provided in the NUREG-1150 study\(^{21} \), and are used to represent the unconditional mean total single-unit accident frequency from all single-unit accident scenarios initiated by internal and external hazards \( (F_{t \text{total}}^s) \). The mean total two-unit accident frequency is then estimated as the product of this unconditional mean total single-unit accident frequency and two other parameters: (1) the global conditional probability of an accident occurring in the co-located unit, given that an accident is assumed to occur in the reference unit (\( \beta \)); and (2) the constant (2), which represents the number of permutations (ordered combinations) for each two-unit accident scenario that could occur, which is two for each two-unit NPP site.
• \( \Sigma_{ij} F_{ij}^t \) = 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.

3.F.4.II. Estimate Adjusted Two-Unit Accident Scenario Risk (Step 9b)

Once the two-unit frequency adjustment factor has been estimated, the adjusted contribution to each selected risk metric from each two-unit accident scenario \( ij \) is estimated using Equation (8):

\[
R^t_{ij} = \alpha^t \cdot (R^t_{ij})_u
\]  

(8)

• \( R^t_{ij} \) = adjusted mean risk contribution to each selected risk metric from the category of two-unit accident scenarios that scenario \( ij \) is assumed to represent. In this context, the term adjusted means the risk contribution has been calibrated using results from previous full-scope NPP Level 3 PRAs to account for the frequency contribution from other two-unit accident scenarios that are not modeled, but that are assumed to result in similar conditional consequence distributions as scenario \( ij \), and therefore are assumed to be represented by scenario \( ij \).

3.F.5. Estimate Total Two-Unit Accident Risk (Step 10)

At the level of the two-unit NPP site, the modeled two-unit accident scenarios are defined to be mutually exclusive. This means that only one of the modeled scenarios can occur at a time at a two-unit site. Under this mutually exclusive condition, the total contribution to each selected risk metric from all two-unit accident scenarios is estimated using Equation (9):

\[
R^t_{total} = \Sigma_{ij} R^t_{ij}
\]  

(9)
• $R_{total}^t = \text{total mean risk contribution to each selected risk metric from all two-unit accident scenarios.}$
3.G. Estimation of the Figure of Merit (Figure 9)


To be able to compare the policy alternatives using the FOM selected to evaluate the effect of expanding the scope of the safety goal policy, each of the selected risk metrics must be estimated under the assumptions for each policy alternative.

Equation (10) is used to calculate the mean value for each selected risk metric under Option 1 (Status Quo), which assumes only the risk contribution from single-unit accident scenarios is included in the calculation:

\[ R_1 = R^S_{\text{total}} \]  

- \( R_1 \) = mean value for each selected risk metric under Option 1 (Status Quo), which assumes only the risk contribution from single-unit accident scenarios is included.

Equation (11) is used to calculate the mean value for each selected risk metric under Option 2, which assumes the risk contributions from both single-unit and two-unit accident scenarios are included in the calculation:

\[ R_2 = R^S_{\text{total}} + R^T_{\text{total}} \]  

- \( R_2 \) = mean value for each selected risk metric under Option 2, which assumes the risk contributions from both single-unit and two-unit accident scenarios are included.
3.G.2. Estimate Safety Goal Quantitative Health Objectives and Margin to Each Objective for Each Policy Alternative (Step 12)

3.G.2.I. Estimate Quantitative Health Objectives Using Data for Selected Year (Step 12a)

In addition to illustrating how the FOM is calculated in Step 13, Figure 9 also illustrates the process and information used to estimate the margin to each safety goal QHO (the early fatality risk QHO and the latent cancer fatality risk QHO) under the assumptions for each policy alternative. This information is used to provide supplementary insights about the effect of expanding the scope of the safety goal policy that are specific to the individual radiological health risk perspective, considering the existing QHOs.

The QHO values that are to be used as the basis for comparison with corresponding results for the average individual early fatality and latent cancer fatality risk metrics, respectively, are calculated using Equations (12) and (13):

\[
QHOF_EF = \left( \frac{1}{1000} \right) \cdot \left( \frac{\text{number of accidental deaths per year}}{\text{US population}} \right) \quad (12)
\]

- \(QHOF_EF\) = quantitative health objective for average individual early fatality risk.

Based on 2013 data from the Centers for Disease Control and Prevention (CDC) National Vital Statistics System\textsuperscript{72} and the U.S. Census Bureau,\textsuperscript{73} the sum of prompt fatality risks resulting from unintentional accidents for the U.S. population is 4.1E-04 per year. This results in an early fatality risk QHO value of 4.1E-07 per year.

\[
QHOF_{LCF} = \left( \frac{1}{1000} \right) \cdot \left( \frac{\text{number of cancer deaths from all causes per year}}{\text{US population}} \right) \quad (13)
\]
• $QHO_{LCF} = \text{quantitative health objective for average individual latent cancer fatality risk. Based on 2013 data from the CDC National Vital Statistics System}^{72} \text{ and the U.S. Census Bureau,}^{73} \text{ the sum of cancer fatality risks resulting from all causes among the U.S. population is 1.9E-03 per year. This results in a latent cancer fatality risk QHO value of 1.9E-06 per year.}$

3.G.2.II. **Estimate Margin to Each Quantitative Health Objective for Each Policy Alternative (Step 12b)**

The margin to each QHO is analogous to the concept of a safety margin used in other contexts and represents the relative distance between the QHO and the value of the corresponding risk metric it is being compared against. In other words, the QHO margin represents the factor by which the calculated risk metric would have to increase to reach the QHO. In practice, this QHO margin is calculated as the ratio of the QHO to the value of the corresponding risk metric. The margin to each QHO is thus calculated for each policy alternative using Equations (14) and (15):

\[
(M_1)_{EF,LCF} = \frac{QHO_{EF,LCF}}{(R_1)_{EF,LCF}} 
\]  

(14)

• $(M_1)_{EF,LCF} = \text{margin to the quantitative health objective for average individual early fatality risk or latent cancer fatality risk under Option 1 (Status Quo), which assumes only the risk contribution from single-unit accident scenarios is included in calculating the value of the corresponding risk metric.}$

• $QHO_{EF,LCF} = \text{quantitative health objective for average individual early fatality risk or latent cancer fatality risk.}$
3.G.3. Estimate Figure of Merit for Each Selected Risk Metric (Step 13)

One FOM was selected to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. This FOM is the relative contribution of multi-unit accident scenarios to the total mean value for each selected risk metric calculated under Option 2, which assumes the contributions from both single-unit and multi-unit accident scenarios are included. This relative contribution is calculated using Equation (16):

\[
FOM = \left( \frac{R_{t,\text{total}}}{R_2} \right) \cdot 100\% = \left( \frac{R_{t,\text{total}}}{R_{t,\text{total}}+R_{t,\text{total}}} \right) \cdot 100\% \quad (16)
\]
Of interest to this dissertation research is whether—for each of the selected risk metrics—the relative contribution of multi-unit accident scenarios to total mean risk is negligible compared to the relative contribution of single-unit accident scenarios. In this context, the term *negligible* relates to the extent to which the impact on the conclusions and risk insights could affect a decision under consideration, with a negligible impact implying that a decision would not be affected.  

For this dissertation research, the relative contribution of multi-unit accident scenarios to total mean risk for each selected risk metric will be judged to be negligible if it is <10%, since this range of values is unlikely to impact a decision with respect to safety goal policy application.
3.H. **Generalization of Methods**

Although these methods have been developed to enable estimation of the contribution to selected risk metrics from single-unit and two-unit accident scenarios for representative two-unit NPP sites, this approach is generalizable and could be applied to estimate the contribution to risk metrics of interest from multi-unit accident scenarios that involve combinations of more than two-units at a shared site that includes more than two co-located units.

Generalization 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 is assumed to occur**. 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 category of multi-unit accident scenarios that have been modeled. The Multiple Greek Letter (MGL) method for treatment of common-cause failure (CCF) events\(^{48}\) could potentially be adapted using operational experience data in the licensee event report (LER) database to develop estimates for applicable conditional probabilities.
• **Potential inter-unit differences in design or operation.** This two-unit case study relied on the key assumption that the two units at each NPP site were identical. This assumption of identical units—which is reasonable for both NPP sites—thereby eliminated the need to develop unique consequence models for all possible permutations (ordered combinations) of two-unit accident scenarios, and reduced the total number of two-unit consequence models that needed to be developed by a factor of two. However, if important differences exist between the design or operation of co-located units, the choice of which unit serves as the reference unit can impact results for cases in which there is a non-zero timing offset between concurrent accident scenarios (i.e., the accident initiation, progression, and radiological release timelines are staggered in time instead of occurring simultaneously). This would require development of unique multi-unit consequence models for each permutation of a multi-unit accident scenario that could occur across multiple units at the NPP site.

The expansion and application of these methods to multi-unit sites comprised of more than two units thus 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 further complicate the analysis. However, existing methods could be adapted to address these additional complexities.
3.I. Chapter Summary

This chapter described the models and analytical tools used to estimate policy analysis metrics and to address the specific aims for this dissertation research. The chapter explained the need for efficient risk estimation models that can be used to calibrate the frequency and conditional consequence results based on SOARCA models using results from previous full-scope NPP Level 3 PRA studies. This calibration is designed to produce approximately equivalent risk results that can then be used to develop meaningful safety goal policy insights, without having to develop a resource-intensive, contemporary full-scope NPP Level 3 PRA. Analytical tools were summarized, with a detailed treatment of the MACCS suite of analytical tools that is used to perform PCA. The chapter delineated and illustrated the process and equations used to estimate: (1) the contribution from single-unit accident scenarios to selected risk metrics; (2) the contribution from two-unit accident scenarios to selected risk metrics; (3) the values of selected risk metrics for each policy alternative; and (4) the value of the FOM used to evaluate the effect of expanding the scope of the safety goal policy to include multi-unit accident scenarios and a broader set of public health risk metrics. Finally, additional consideration was given to how these methods—which can be directly applied to NPP sites comprised of two co-located operating reactor units—can be generalized for application to NPP sites comprised of more than two units.

Chapter 4 will present the results of the policy analysis and its policy implications.
Chapter 4. Results and Policy Implications

4.A. Chapter Introduction and Overview

Chapter 1 introduced central concepts relevant to the U.S. Nuclear Regulatory
Commission (USNRC) safety goal policy and probabilistic analysis techniques used
to measure safety goal attainment, as well as gaps in the safety goal policy that
motivate this dissertation research. It further described principal design aspects of a
policy analysis that has been performed to evaluate the effects of a hypothetical
expansion in the scope of the safety goal policy to address these limitations. Based
on a review of the existing literature, Chapter 2 then provided a more detailed
description of the background information that is germane to this dissertation
research. Chapter 3 described the models and analytical tools used to estimate policy
analysis metrics and to address the specific aims for this dissertation research. It
specified efficient risk estimation models that can be used to calibrate the frequency
and conditional consequence results based on models from the State-of-the-Art
Reactor Consequence Analyses (SOARCA) Project using results from previous
full-scope nuclear power plant (NPP) Level 3 probabilistic risk analysis (PRA)
studies. This calibration is designed to estimate approximately equivalent risk
results that can then be used to develop meaningful safety goal policy insights,
without having to develop a resource-intensive, contemporary full-scope NPP Level 3
PRA.

This chapter presents the results of the policy analysis and its policy implications. It
first provides an overview of how to interpret risk results from NPP Level 3 PRA
studies. Next, results for selected risk metrics are summarized for both single-unit
and two-unit accident scenarios under base case analysis assumptions for each of the two representative nuclear power plant (NPP) sites selected as the study population for this dissertation research. Results for the figure of merit (FOM) used to evaluate the effect of the hypothetical expansion in the scope of the safety goal policy are then presented and interpreted for three different risk perspectives represented by the selected risk metrics: (1) individual radiological health risk perspective; (2) societal radiological health risk perspective; and (3) societal non-radiological health risk perspective. Finally, results and insights from one-way sensitivity analyses designed to evaluate the effect of variation in two factors are presented: (1) the assumed level of dependence between co-located reactor units, which impacts the frequency of modeled two-unit accident scenarios; and (2) the assumed timing offset (delay time) between concurrent accident scenarios involving co-located reactor units, which impacts the conditional consequences of modeled two-unit accident scenarios.
4.B. Presentation of Risk Results

This section provides a general overview of risk results from NPP Level 3 PRA studies and the types of risk results that will be presented and interpreted to develop risk-based safety goal policy insights for this dissertation research.

4.B.1. Risk Results from Nuclear Power Plant Level 3 Probabilistic Risk Analysis Studies

The principal output of a NPP Level 3 PRA study is a quantitative characterization of the risks to the public that are attributable to a broad spectrum of potential accident scenarios involving a NPP of interest. These public risks are typically displayed using risk curves (also termed exceedance frequency curves) that illustrate the frequency (or probability of frequency if an integrated uncertainty analysis is performed) of exceeding different consequence levels for a specified set of consequence metrics. Historically, NPP Level 3 PRA studies have focused on the risk of various offsite radiological consequences, including: (1) radiological dose to the affected population; (2) average individual risk of experiencing radiological health effects; (3) total numbers of radiological health effect cases; (4) land areas contaminated at different levels of radioactivity; and (5) economic costs. These frequency (or probability of frequency) distributions are typically summarized using various statistical summary measures in the presentation and interpretation of results. The most common summary measures used in NPP Level 3 PRA studies include the: (1) mean (expected value); (2) median (50th percentile); (3) 95th percentile; and (4) 5th percentile.
4.B.2. Risk Results Used in This Dissertation Research

The USNRC decided during implementation of the safety goal policy that mean values from such PRA studies are to be used for making safety-related decisions for safety goal policy applications involving NPPs. For this reason, this dissertation research presents and interprets results for only the mean values of selected risk metrics when relying on the use of statistical summary measures, including in the calculation, presentation, and interpretation of results for the FOM used to evaluate the effects of a hypothetical expansion in the scope of the safety goal policy to include the contribution from multi-unit accident scenarios.

However, for the base case analysis, these mean value summary measures and the FOM for each selected risk metric will then be supplemented with three further ways to characterize risk to obtain additional risk-based safety goal policy insights:

1. **Scenario-specific frequency and conditional consequence results for each representative NPP site.** Tables that include the mean frequency and mean conditional consequence values for all modeled single-unit and two-unit accident scenarios used to calculate total mean risk and the FOM for each selected risk metric are presented in Appendix II for the representative boiling-water reactor (BWR) site (Peach Bottom) and in Appendix III for the representative pressurized-water reactor (PWR) site (Surry). These tables include information at the level of each modeled accident scenario and provide supporting documentation that can be used to: (1) examine tradeoffs between accident scenario frequencies and conditional consequences; and (2) facilitate review and verification of calculated results.
2. **Risk profiles for both representative NPP sites by risk metric and safety goal policy alternative.** Appendix IV presents site-specific risk profiles that illustrate the relative contributions of all modeled single-unit and two-unit accident scenarios to the total mean value of each selected risk metric and safety goal policy alternative. These risk profiles facilitate identification of: (1) categories of accident scenarios that contribute significantly to each selected risk metric; (2) how the relative importance of different categories of accident scenarios differs by safety goal policy alternative—which is driven by the effect of including the risk contribution from two-unit accident scenarios; and (3) how the relative importance of different categories of accident scenarios and the effect of including two-unit accident scenarios differs by NPP site.

3. **Risk curves for both representative NPP sites by risk metric and safety goal policy alternative.** Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified consequence levels for each selected risk metric, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative. These risk curves illustrate the variability in risk results arising from random processes that influence two elements of the risk triplet for each accident scenario: (1) frequency (or probability of frequency if an integrated uncertainty analysis is performed)—which depends primarily on the combinations of events that give rise to the accident scenario; and (2) conditional consequences—which depend primarily on the potential weather conditions that can exist at the time an accident scenario occurs. Although mean values are used for making safety-related decisions—which is why they are used in calculating the FOM for this dissertation research—they are difficult to
interpret for random variables that are not dichotomous, which go beyond whether an event occurs or does not occur to account for different levels of severity for events that occur within a specified time-period. For these types of random variables in PRA studies, risk curves provide information that is more intuitive and easier to understand. Since all of the conditional consequence measures associated with the set of risk metrics selected for this dissertation research are considered non-dichotomous random variables, risk curves will be used to interpret results and to provide additional safety goal policy insights that—while not central to the specific aims of this dissertation research—go beyond those that can be derived from only the FOM and total mean risk values. These additional insights will be used to help determine the significance of research findings and will illuminate issues for future research.
4.C. Base Case Analysis: Effect of Including Contribution from Multi-Unit Accident Scenarios to Selected Risk Metrics (Aim 1)

This section presents and interprets results for base case analyses that were performed to evaluate the effect of including the risk contribution from multi-unit accident scenarios to selected risk metrics for both representative two-unit NPP sites selected as case studies for this dissertation research. These base case analyses relied on two assumptions that respectively influence the frequency and conditional consequence elements of the risk triplet for two-unit accident scenarios: (1) the level of dependence between co-located reactor units was assumed to be 0.1, which implies a 10% chance of a co-located unit experiencing a concurrent accident scenario, given that a single-unit accident scenario involving any unit occurs at the site; and (2) the timing offset (delay time) between concurrent accident scenarios involving co-located units that comprise each two-unit accident scenario was assumed to be zero, which implies the concurrent accident scenarios occur simultaneously.

4.C.1. Summary Results and Policy Implications

This section: (1) summarizes results and policy implications based on the FOM, supplemented with insights derived from comparison of results for average individual early fatality risk and average individual latent cancer fatality risk to corresponding values for existing quantitative health objectives (QHOs); and (2) describes the need for a more detailed treatment of risk results to obtain a more complete characterization of risk for each public health perspective and additional risk-based safety goal policy insights.
4.C.1.I. Figure of Merit Results and Policy Implications

Table VII illustrates summary results by risk metric for the representative BWR site (Peach Bottom) under base case analysis assumptions. As shown, the FOM—the relative contribution of two-unit accident scenarios to the total mean value for each selected risk metric calculated under Option 2, which assumes the risk contributions from both single-unit and two-unit accident scenarios are included—ranges from 23% to 64% for the representative BWR site across all selected risk metrics. Although the relative importance of two-unit accident scenarios varies by risk metric—with two-unit accident scenarios being more important to risk metrics related to early fatality risk—the relative contribution of two-unit accident scenarios to total mean risk is more than 10% across all risk metrics, and is therefore considered a non-negligible contributor to total mean risk. Two-unit accident scenarios thus represent a significant risk contributor for the representative BWR site across the range of selected risk metrics.

Table VIII illustrates summary results by risk metric for the representative PWR site (Surry) under base case analysis assumptions. As shown, the FOM ranges from 17% to 29% for the representative PWR site. Although the relative importance of two-unit accident scenarios still varies to some extent by risk metric, the relative contribution of two-unit accident scenarios to total mean risk is more than 10% across all risk metrics, and is therefore considered a non-negligible contributor to total mean risk. Two-unit accident scenarios thus also represent a significant risk contributor for the representative PWR site across the range of selected risk metrics.
### Table VII. Summary Results for Representative BWR Site (Peach Bottom)

<table>
<thead>
<tr>
<th>Risk Metric</th>
<th>Accident Scenario Risk Contribution</th>
<th>Option 1a</th>
<th>Option 2b</th>
<th>Figure of Merit (FOM)c</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Single-Unit (per year)</td>
<td>Two-Unit (per year)</td>
<td>Total Mean Risk (per year)</td>
<td>Total Mean Risk (per year)</td>
</tr>
<tr>
<td><strong>Individual Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Average Individual Early Fatality Risk (0-1 mile)</td>
<td>6.5E-13</td>
<td>1.0E-12</td>
<td>6.5E-13</td>
<td>1.6E-12</td>
</tr>
<tr>
<td>Average Individual Latent Cancer Fatality Risk (0-10 miles)</td>
<td>2.8E-09</td>
<td>8.1E-10</td>
<td>2.8E-09</td>
<td>3.6E-09</td>
</tr>
<tr>
<td><strong>Societal Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Early Fatality Risk (0-50 miles)</td>
<td>3.1E-11</td>
<td>5.4E-11</td>
<td>3.1E-11</td>
<td>8.5E-11</td>
</tr>
<tr>
<td>Total Latent Cancer Fatality Risk (0-50 miles)</td>
<td>6.3E-03</td>
<td>2.3E-03</td>
<td>6.3E-03</td>
<td>8.6E-03</td>
</tr>
<tr>
<td><strong>Societal Non-Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Emergency Phase Population Relocation Risk (0-50 miles)</td>
<td>6.9E+00</td>
<td>2.1E+00</td>
<td>6.9E+00</td>
<td>9.0E+00</td>
</tr>
<tr>
<td>Total Recovery Phase Population Relocation Risk (0-50 miles)</td>
<td>1.7E+00</td>
<td>5.7E-01</td>
<td>1.7E+00</td>
<td>2.3E+00</td>
</tr>
</tbody>
</table>

- ^a^ For Option 1, which assumes only the contribution from single-unit accident scenarios is included, the total mean risk for each selected risk metric is equal to the single-unit accident scenario risk contribution.
- ^b^ For Option 2, which assumes the contributions from both single-unit and two-unit accident scenarios are included, the total mean risk for each selected risk metric is calculated by summing the single-unit and two-unit accident scenario risk contributions.
- ^c^ The FOM is the relative contribution of two-unit accident scenarios to total mean risk under Option 2. This FOM is calculated for each selected risk metric by dividing the two-unit accident scenario risk contribution by the Option 2 total mean risk and multiplying by 100%.
Table VIII. Summary Results for Representative PWR Site (Surry)

<table>
<thead>
<tr>
<th>Risk Metric</th>
<th>Accident Scenario Risk Contribution</th>
<th>Option 1&lt;sup&gt;a&lt;/sup&gt; Total Mean Risk (per year)</th>
<th>Option 2&lt;sup&gt;b&lt;/sup&gt; Total Mean Risk (per year)</th>
<th>Figure of Merit (FOM)&lt;sup&gt;c&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Individual Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Average Individual Early Fatality Risk (0-1 mile)</td>
<td>1.8E-11</td>
<td>6.9E-12</td>
<td>1.8E-11</td>
<td>2.4E-11</td>
</tr>
<tr>
<td>Average Individual Latent Cancer Fatality Risk (0-10 miles)</td>
<td>4.3E-09</td>
<td>1.6E-09</td>
<td>4.3E-09</td>
<td>5.9E-09</td>
</tr>
<tr>
<td><strong>Societal Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Early Fatality Risk (0-50 miles)</td>
<td>7.0E-10</td>
<td>2.9E-10</td>
<td>7.0E-10</td>
<td>9.9E-10</td>
</tr>
<tr>
<td>Total Latent Cancer Fatality Risk (0-50 miles)</td>
<td>1.7E-03</td>
<td>6.5E-04</td>
<td>1.7E-03</td>
<td>2.4E-03</td>
</tr>
<tr>
<td><strong>Societal Non-Radiological Health Risk Perspective</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Total Emergency Phase Population Relocation Risk (0-50 miles)</td>
<td>1.4E+01</td>
<td>2.8E+00</td>
<td>1.4E+01</td>
<td>1.7E+01</td>
</tr>
<tr>
<td>Total Recovery Phase Population Relocation Risk (0-50 miles)</td>
<td>1.2E-02</td>
<td>4.8E-03</td>
<td>1.2E-02</td>
<td>1.7E-02</td>
</tr>
</tbody>
</table>

<sup>a</sup> For Option 1, which assumes only the contribution from single-unit accident scenarios is included, the total mean risk for each selected risk metric is equal to the single-unit accident scenario risk contribution.

<sup>b</sup> For Option 2, which assumes the contributions from both single-unit and two-unit accident scenarios are included, the total mean risk for each selected risk metric is calculated by summing the single-unit and two-unit accident scenario risk contributions.

<sup>c</sup> The FOM is the relative contribution of two-unit accident scenarios to total mean risk under Option 2. This FOM is calculated for each selected risk metric by dividing the two-unit accident scenario risk contribution by the Option 2 total mean risk and multiplying by 100%. 

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Figure 10 illustrates both the relative and absolute contributions of single-unit and two-unit accident scenarios to total mean risk for each representative NPP site and each risk metric under Option 2. Figure 10 provides a visual representation of the information provided in Table VII and Table VIII and further shows that single-unit accident scenarios contribute most to the total mean value for most selected risk metrics, comprising more than 70% of total mean risk in all but two cases. The two exceptions are the risk metrics related to early fatality risk for the representative BWR site: (1) average individual early fatality risk within one mile of the site; and (2) total early fatality risk within 50 miles of the site. For these exceptions, single-unit accident scenarios comprise 39% and 36% of total mean risk, respectively. This unique finding will be described in more detail in section 4.C.1., which focuses on the individual radiological health risk perspective.

**Key Finding 1: Including the risk contribution from multi-unit accident scenarios results in a non-negligible increase in the total mean value for all selected risk metrics.**

A key finding from the summary results presented in Table VII, Table VIII, and Figure 10 is that—for all risk metrics selected to represent a diverse set of public health perspectives—two-unit accident scenarios comprise more than 10% of the total mean risk under Option 2. This means that if the scope of the safety goal policy was to be expanded to include the risk contribution from multi-unit accident scenarios, this contribution—at least for the representative two-unit NPP sites evaluated as case studies for this dissertation research—would not be considered negligible. Put another way, expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios could result in a substantial
enough increase in total mean risk to impact safety-related decisions for safety goal policy applications. Moreover, this finding is not dependent upon whether the scope of the safety goal policy and QHOs are also expanded to include measures of societal risk for radiological health and non-radiological health consequences.

Figure 10. Contributions of Single-Unit and Two-Unit Accident Scenarios to Total Mean Risk for Each Nuclear Power Plant Site and Each Risk Metric Under Option 2. Relative contributions are illustrated by the width of the corresponding bar. Absolute contributions are illustrated by the numerical values embedded within each bar. Single-unit accident scenarios contribute most to the total mean value for most selected risk metrics, comprising more than 70% of total mean risk for Option 2 for all but two risk metrics for the representative BWR site: (1) average individual early fatality risk within one mile; and (2) total early fatality risk within 50 miles. Although single-unit accident scenarios generally contribute most to total mean risk, the contribution from two-unit accident scenarios under Option 2 is more than 10% for all risk metrics, and thus would not be considered negligible if the safety goal policy was to be expanded to include multi-unit accident scenarios—even if the safety goal policy and QHOs are not also expanded to include measures of societal risk for radiological health and non-radiological health consequences.
4.C.1.II. Quantitative Health Objective Results and Policy Implications

The USNRC safety goal policy specifies two high-level qualitative safety goals pertaining to commercial NPP operations: (1) an individual risk safety goal specifying in part that individuals should bear no significant additional risk to life and health; and (2) a societal risk safety goal specifying in part that societal risks to life and health should be comparable to or less than risks of generating electricity by viable competing technologies. Each of these qualitative safety goals are supported by a corresponding QHO that can be used to determine whether and to what extent the safety goals have been achieved. These QHOs used to measure safety goal attainment were developed for comparison with corresponding results for average individual early fatality and latent cancer fatality risk metrics. One QHO-based measure of safety goal attainment is the margin to each QHO. The QHO margin is analogous to the concept of a safety margin used in other contexts and represents the relative distance between the QHO and the value of the corresponding risk metric it is being compared against. In other words, the QHO margin represents the factor by which the calculated risk metric would have to increase to reach the QHO. In practice, this QHO margin is calculated as the ratio of the QHO to the mean value of the corresponding risk metric.

In addition to the FOM results presented in section 4.C.1.I., this section presents results for the margin to both the early fatality risk QHO and the latent cancer fatality risk QHO under the base case analysis assumptions for each policy alternative. These results provide supplementary insights about the effect of a hypothetical expansion in the scope of the safety goal policy that are specific to the individual radiological health risk perspective, considering the existing QHOs.
Equations for calculating the existing QHO values and the margin to each QHO are provided in section 3.G.2. of Chapter 3. Based on 2013 data from the Centers for Disease Control and Prevention (CDC) National Vital Statistics System\textsuperscript{72} and the U.S. Census Bureau\textsuperscript{73}: (1) the sum of prompt fatality risks resulting from unintentional accidents for the U.S. population is 4.1E-04 per year, which results in an early fatality risk QHO value of 4.1E-07 per year using Equation (12); and (2) the sum of cancer fatality risks resulting from all causes among the U.S. population is 1.9E-03 per year, which results in a latent cancer fatality risk QHO value of 1.9E-06 per year using Equation (13). Applying these QHO values and the total mean values for average individual early fatality risk within one mile and average individual latent cancer fatality risk within ten miles, the margin to each QHO for each representative NPP site and each safety goal policy alternative was calculated under base case analysis assumptions using Equation (14) and Equation (15), respectively. The results are displayed in Figure 11.

**Key Finding 2: Including the risk contribution from multi-unit accident scenarios to existing safety goal QHO risk metrics would likely not impact safety-related decisions for safety goal policy applications.**
Figure 11. Margin to Each Quantitative Health Objective for Each Representative NPP Site and Each Safety Goal Policy Alternative. Although including the risk contribution from two-unit accident scenarios reduces the margin to each QHO for each representative NPP site: (1) more than four orders of magnitude in margin remain to the early fatality risk QHO for each site; and (2) more than two orders of magnitude in margin remain to the latent cancer fatality risk QHO for each site.

Figure 11 shows that although including the risk contribution from two-unit accident scenarios reduces the margin to each QHO for each representative NPP site: (1) more than four orders of magnitude in margin remain to the early fatality risk QHO for each site; and (2) more than two orders of magnitude in margin remain to the latent cancer fatality risk QHO for each site. This key finding suggests that—
even when the risk contribution from multi-unit accident scenarios is included—the total mean values of the risk metrics for individual radiological health effects would have to increase by several orders of magnitude for the corresponding safety goals to no longer be satisfied. The related policy implication is that—assuming the QHOs continue to be limited to existing measures of individual risk of experiencing radiological health effects—expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios would likely not impact safety-related decisions for safety goal policy applications. This is because—from an individual radiological health risk perspective—these results indicate the residual risk is already at an acceptably low level, which implies the limited resources that would be applied to a proposed regulatory action could be better applied to alternative courses of action (e.g., other activities that support USNRC’s statutory responsibility to ensure the public is adequately protected from hazards posed by civilian use of nuclear materials, voluntary nuclear industry initiatives, or even activities to reduce risks of non-nuclear hazards to public health and safety).

4.C.1.III. The Need for More Complete Risk Characterizations and Additional Risk-Based Safety Goal Policy Insights

The overall aim for this dissertation research includes two elements: (1) 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 for multi-unit NPP sites; and (2) to evaluate the effect of expanding the scope of the safety goal policy to include consideration of a broader set of public health risk metrics that includes measures of societal risk for both radiological and non-radiological health consequences.
The summary results presented in sections 4.C.1.I and 4.C.1.II primarily provide insight into the first element, with two key findings that appear to be in tension. On one hand, the FOM results presented in Table VII, Table VIII, and Figure 10 clearly illustrate that—for the base case analysis—expanding the scope of the safety goal policy to include consideration of the risk contribution from multi-unit accident scenarios results in a non-negligible increase in the total mean value for all risk metrics selected to represent a broad set of public health perspectives; multi-unit accident scenarios thus constitute a significant contributor to total accident risk for both representative NPP sites across a range of risk metrics. On the other hand, the QHO margin results presented in Figure 11 clearly illustrate that—also for the base case analysis—several orders of magnitude in margin to each QHO remain for each representative NPP site, even when the risk contribution from multi-unit accident scenarios is included; this implies that expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios would likely not impact safety-related decisions for safety goal policy applications. These two findings collectively suggest that—although multi-unit accident scenarios may be a significant source of accident risk for each representative NPP site—the level of residual risk to the public is already at an acceptably low level, with sufficient margin to each QHO to provide assurance that the residual risk level would remain acceptable even if multi-unit accident scenarios were considered; there would thus be no practical benefit to expanding the scope of the safety goal policy to include multi-unit accident scenarios, which would necessarily increase the resources needed to perform supporting NPP PRAs.
However, these summary results do not sufficiently illuminate the second element of the overall aim for this dissertation research. Beyond examining the FOM results to determine whether the contribution from two-unit accident scenarios to total mean risk for each selected risk metric is negligible, another important consideration that is central to the second element is whether additional risk insights can be obtained by expanding the scope of the safety goal policy and QHOs beyond the individual radiological health risk perspective to include the societal radiological and non-radiological health risk perspectives. In other words, another objective of this dissertation research is to identify whether potentially important risk insights related to public health and safety could be missed by limiting the scope of the safety goal policy to the existing set of QHO risk metrics that is limited to measures of individual risk of experiencing radiological health effects. To answer this question, different ways of characterizing risk will be used to provide a more complete characterization of the risks to the public that are attributable to potential single-unit and two-unit accident scenarios involving both representative NPP sites.

Although this information cannot be used to determine whether these risks are acceptable—since quantitative objectives do not yet exist for comparison against the societal risk measures—it can provide a valuable input to current USNRC and nuclear industry stakeholder deliberations about whether and to what extent the safety goal policy should be expanded to include QHOs that address societal risk.22,28

To obtain these additional insights into the second element, a more detailed evaluation of the base case analysis results for each public health perspective addressed in this dissertation research is provided in sections 4.C.2., 4.C.3., and 4.C.4. These sections draw upon additional risk results presented and displayed in
the following appendices to this dissertation: (1) Appendix II—Results Tables for the Representative Boiling-Water Reactor Site (Peach Bottom); (2) Appendix III—Results Tables for the Representative Pressurized-Water Reactor Site (Surry); (3) Appendix IV—Risk Profiles for Both Representative Nuclear Power Plant (NPP) Sites by Risk Metric and Safety Goal Policy Alternative; and (4) Appendix V—Risk Curves for Both Representative Nuclear Power Plant (NPP) Sites by Risk Metric and Safety Goal Policy Alternative.

4.C.2. Individual Radiological Health Risk Perspective

This section describes additional insights derived from a more detailed evaluation of the base case analysis results for the individual radiological health risk perspective.

4.C.2.I. Average Individual Early Fatality Risk (0-1 mile)

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-I in Appendix II and Table III-I in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted risk values for average individual early fatality risk within one mile of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results yields three supplementary insights.

Supplementary Insight 1: Categories of accident scenarios involving significant releases of Iodine-131 (I-131) dominate average individual early fatality risk within one mile of each representative NPP site.
Table III in Chapter 1 summarizes various characteristics of the single-unit accident scenarios modeled as part of the SOARCA pilot study that were used to create two-unit accident scenarios for evaluation as part of this dissertation research. One of these characteristics is the I-131 release fraction, which represents the percentage of I-131 stored in the reactor core that is released to the environment due to the accident scenario. I-131 is preferentially absorbed in the thyroid gland and previous studies have shown that release categories involving relatively large amounts of I-131 are significant contributors to early fatality risk.\textsuperscript{19-21}

Table III in Chapter 1 shows that two single-unit accident scenarios result in release of more than 10% of the I-131 stored in the reactor core: (1) Short-Term Station Blackout–Base Case (STSBO-Base), which has an I-131 release fraction of 12% for the representative BWR site; and (2) Interfacing Systems Loss-of-Coolant Accident (ISLOCA), which has an I-131 release fraction of 16% for the representative PWR site. Table II-I in Appendix II and Table III-I in Appendix III show that categories of single-unit and two-unit accident scenarios that include either of these accident scenarios that result in a significant release of I-131 completely dominate average individual early fatality risk within one mile of each representative site. This finding is not dependent upon whether the risk contribution from multi-unit accident scenarios is included in the total mean risk calculation, and is consistent with findings from previous studies.\textsuperscript{19-21}

**Supplementary Insight 2:** Frequency-conditional consequence tradeoffs for single-unit versus two-unit accident scenarios differ between the representative NPP sites with respect to average individual early fatality risk within one mile.
Based on how it is calculated, there are two factors that influence the risk contribution from two-unit accident scenarios and its relative contribution to total mean risk when considered with the contribution from single-unit accident scenarios. The first factor is the tradeoff between the frequency and conditional consequences of two-unit accident scenarios compared with their constituent single-unit accident scenarios. Two-unit accident scenarios are expected to result in conditional consequences that are greater than or equal to those of their constituent single-unit accident scenarios. However, given that multiple additional failure events are required to jointly occur for a two-unit accident scenario, the frequencies of two-unit accident scenarios are expected to be less than those of single-unit accident scenarios. The relative contribution of two-unit accident scenarios to a specified risk metric will thus depend upon how the relative reduction in frequency compares with the relative increase in conditional consequences expected for two-unit accident scenarios compared with single-unit accident scenarios. The second factor that influences the risk contribution of two-unit accident scenarios is the number of two-unit accident scenarios that have non-zero frequency and non-zero conditional consequences and that thus contribute to the risk metric of interest. In other words, for a given level of scenario-specific risk above zero, as the number of scenarios that contribute to risk increases, the total risk increases.

For the representative BWR site, Table II-I shows that the conditional consequence results for all two-unit accident scenarios that result in non-zero consequences are all greater than those for the single-unit accident scenario that results in non-zero consequences (STSBO-Base). In fact, the BWR5 accident scenario comprised of STSBO-Base scenarios in both the reference unit and co-located unit results in
conditional consequences that are 40 times greater than those of the single-unit STSBO-Base scenario. Although the frequencies of these two-unit accident scenarios are generally lower than the frequency of the single-unit STSBO-Base scenario by a factor that is greater than the corresponding increase in conditional consequences, two-unit accident scenarios appear to contribute more to total mean risk under Option 2 than single-unit accident scenarios for the representative BWR site for two reasons: (1) the frequency-conditional consequence tradeoff is close (within an order of magnitude) for most two-unit accident scenarios compared with the single-unit STSBO-Base scenario (i.e., the relative increase in conditional consequences is almost offset by the relative reduction in frequency); and (2) there are five times as many two-unit accident scenarios that contribute to average individual early fatality risk than there are single-unit accident scenarios (i.e., there is only one single-unit STSBO-Base scenario compared with five two-unit accident scenarios that include the STSBO-Base scenario as a constituent).

By contrast, Table III-1 shows a different pattern for the representative PWR site. Like the representative BWR site, the conditional consequence results for all two-unit accident scenarios that result in non-zero consequences are all greater than those for the single-unit accident scenario that results in non-zero consequences (ISLOCA). However, all but one of these two-unit accident scenarios result in conditional consequences that are identical to those of the single-unit accident scenarios, which suggests that the ISLOCA scenario is the limiting scenario that drives early fatality risk when it occurs concurrently with other accident scenarios that do not release as much I-131; the exception is the two-unit accident scenario that involves concurrent ISLOCA scenarios in both the reference unit and co-located
unit (PWR16), which results in conditional consequences that are about four times
greater than those of the single-unit ISLOCA scenario. Two-unit accident scenarios
thus have a modest impact on the conditional consequence element of early fatality
risk when compared to single-unit accident scenarios. However, analysis of the data
in Table III-1 reveals that the frequencies of most two-unit accident scenarios with
non-zero conditional consequences are 1-3 orders of magnitude lower than those of
the single-unit ISLOCA scenario. Since the conditional consequence element is
approximately the same, this means the contribution to the total mean value of
average individual early fatality risk from most two-unit accident scenarios is
negligible compared to the risk contribution from the single-unit ISLOCA scenario,
with only two two-unit scenarios that include an ISLOCA coupled with a Long-Term
Station Blackout (LTSBO) scenario (PWR4 and PWR13) contributing significantly to
total mean risk.

These insights together provide a more complete explanation for the FOM results
reported in Table VII and Table VIII, and provide an additional view into why the
risk contribution from two-unit accident scenarios to average individual early
fatality risk within one mile is greater for the representative BWR site (FOM=61%)
than for the representative PWR site (FOM=28%).

**Supplementary Insight 3: Relying solely on risk insights for single-unit
accident scenarios can lead to flawed risk management strategies for
reducing average individual early fatality risk within one mile.**

In addition to the FOM results that illustrate two-unit accident scenarios are non-
negligible contributors to the total mean value for average individual early fatality
risk within one mile, further support for expanding the scope of the safety goal policy to include the contribution from multi-unit accident scenarios is obtained by examining the categories of accident scenarios that result in early fatalities in Table III-I. As stated previously with Supplementary Insight 1, single-unit and two-unit accident scenarios that include accident scenarios that result in a significant release of I-131 (the STSBO-Base scenario for the representative BWR site or the ISLOCA scenario for the representative PWR site) completely dominate average individual early fatality risk within one mile of each representative site. Table III-I shows that: (1) the ISLOCA scenario is the only single-unit accident scenario to contribute to average individual early fatality risk within one mile; and (2) all two-unit accident scenarios that include ISLOCA as a constituent also contribute to early fatality risk.

However, there is an additional two-unit accident scenario that results in early fatalities, but contributes less than 1% to the total mean value for average individual early fatality risk within one mile. This scenario involves concurrent Short-Term Station Blackout with Thermally-Induced Steam Generator Tube Rupture (STSBO-TISGTR) scenarios in both the reference unit and co-located unit (PWR11). Table III in Chapter 1 shows that the STSBO-TISGTR is the most rapidly progressing accident scenario, with only three hours from accident initiation to the onset of core damage, and only 0.5 hours from onset of core damage to release of radiological materials to the environment. This accident scenario thus allows little time for offsite response organizations (OROs) to implement emergency response protective actions (e.g., evacuation, sheltering, dose-dependent relocation) among the affected population. Although this accident scenario progresses rapidly and affords relatively little time for implementing protective actions, only a relatively small
amount of I-131 (1% of core inventory) is released due to the accident. It is likely for this reason that the single-unit STSBO-TISGTR does not result in early fatalities and therefore does not contribute to average individual early fatality risk within one mile.

Early fatalities are a type of deterministic radiological health effect described in section 2.E.1. of Chapter 2. For an early fatality to occur, many cells must die to cause sufficient damage to tissues. This typically requires high doses of radiation over short periods of time to exceed dose thresholds for exposed tissues. Once threshold doses for exposed tissues are exceeded, deterministic health effects—including early fatalities—are predicted to occur with certainty, with the severity of the effect increasing as the dose level increases. A plausible explanation for why the single-unit STSBO-TISGTR does not result in early fatalities, but the two-unit accident scenario involving concurrent STSBO-TISGTR scenarios in both the reference unit and co-located unit (PWR11) does result in early fatalities, is that the combined timing and amount of radiological release results in sufficiently large radiological doses among the affected population to exceed the threshold dose for early fatalities for a subset of the exposed population.

This threshold effect highlights an important risk management issue when considering whether to expand the scope of the safety goal policy to include the contribution from multi-unit accident scenarios. A simple thought experiment using the results in Table III-I will suffice to illustrate the issue. Assume only the information about the single-unit accident scenarios is available. This information suggests that implementation of a hypothetical risk reduction measure that eliminates the category of accident scenarios represented by the ISLOCA scenario
would eliminate early fatality risk within one mile of the representative PWR site. However, since the two-unit accident scenario involving concurrent STSBO-TISGTR scenarios in both the reference unit and co-located unit (PWR11) results in early fatalities and thus contributes to early fatality risk within one mile, this risk management strategy that relies solely on insights from the single-unit accident scenarios is clearly flawed. One could argue that the PWR11 scenario is a negligible contributor to early fatality risk, and therefore conclude that early fatality risk within one mile would be practically eliminated by such a hypothetical risk management strategy. However, this finding illustrates a potential pitfall of excluding the risk contribution from multi-unit accident scenarios—especially for risk metrics related to deterministic effects that require exceedance of some threshold level for the effects to be observed. While single-unit accident scenarios may not be sufficient to cause threshold levels to be exceeded, concurrent accident scenarios involving two or more co-located units may be sufficient to do so.

**Insights from Site-Specific Risk Profiles (Appendix IV)**

Figure IV-1 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for average individual early fatality risk within one mile of each representative NPP site for each safety goal policy alternative. The results presented in Figure IV-1 further reinforce the finding that categories of accident scenarios involving significant releases of I-131 dominate average individual early fatality risk within one mile of both sites.
Insights from Site-Specific Risk Curves (Appendix V)

Figure V-1 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified levels of average individual early fatality risk within one mile of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative. Evaluation of Figure V-1, together with the remaining figures in Appendix V, yields one additional supplementary insight.

Supplementary Insight 4: Including the contribution from two-unit accident scenarios shifts the risk curve: (1) up, which means that for a specified consequence level, the frequency of exceeding that consequence level increases; and/or (2) to the right, which means the maximum consequence level observed across all accident scenarios and all weather trials increases.

As shown in Figure V-1, including the contribution from two-unit accident scenarios (i.e., comparing the risk curve for Option 1 to the risk curve for Option 2) has one or both of the following effects in the region of higher levels of average individual early fatality risk: (1) it shifts the risk curve up, which means that for a specified level of average individual early fatality risk, the frequency of exceeding that average individual early fatality risk level increases; and/or (2) it extends the risk curve to the right, which means the maximum average individual early fatality risk level observed across all accident scenarios and all weather trials increases—with lower-frequency, higher-consequence two-unit accident scenarios giving rise to the extreme right tail of the Option 2 risk curve. Examination of Figure V-1 through Figure V-6 reveals that this finding broadly applies across all selected risk metrics. For
remaining risk metrics, the description of insights from site-specific risk curves will thus focus on the examination of specific points on each risk curve, as described in the following paragraphs.

To provide additional context for interpreting the results displayed in Figure V-1, specific points on the Option 1 and Option 2 risk curves are examined. This information: (1) illustrates one effect of including the risk contribution from two-unit accident scenarios; and (2) can be used to inform potential future decisions regarding the acceptability of these results. The point of interest varies by risk metric and typically corresponds to a benchmark consequence level that can be used to put the observed results into a familiar and understandable perspective. One measure of the effect of including the risk contribution from two-unit accident scenarios is the change in the mean exceedance frequency for this consequence level of interest when comparing the risk curve for Option 2 to Option 1.

For average individual early fatality risk within one mile of each representative NPP site, the risk level of interest is 1E-06, which means that individuals residing within one mile of each site have at least a 1 in 1,000,000 chance of dying from acute radiation exposure if an accidental radiological release from the site occurs. This risk level is selected because it is commonly used for expressing individual risks and is more than two orders of magnitude below the background risk of accidental death that individuals in the US are generally exposed to in a given year, based on 2013 data.\textsuperscript{72}

For the representative BWR site (Panel I), the mean frequency of exceeding an average individual early fatality risk level of 1E-06 is 1E-07 per year for Option 1
and 2E-07 per year for Option 2. This means that an accidental release from the representative BWR site that results in an average individual early fatality risk level of greater than 1E-06 within one mile is expected to occur about once every 10 million years if only single-unit accident scenarios are considered, and once every 5 million years if both single-unit and two-unit accident scenarios are considered.

For the representative PWR site (Panel II), the mean frequency of exceeding an average individual early fatality risk level of 1E-06 is 7E-08 per year for Option 1 and 1E-07 per year for Option 2. This means that an accidental release from the representative PWR site that results in an average individual early fatality risk level of greater than 1E-06 within one mile is expected to occur about once every 14 million years if only single-unit accident scenarios are considered, and once every 10 million years if both single-unit and two-unit accident scenarios are considered.

To put these results in perspective, a very large asteroid capable of causing a global catastrophe is expected to occur about once every 50 million years, with a 90% confidence interval that ranges from a frequency of once every 20 million years to once every 1 billion years. These results suggest that—based on the assumptions made in the base case analysis for this dissertation research—the magnitude of average individual early fatality risks within one mile of each representative NPP site is extremely low.
4.C.2.II. Average Individual Latent Cancer Fatality Risk (0-10 miles)

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-II in Appendix II and Table III-II in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted risk values for average individual latent cancer fatality risk within ten miles of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results yields one additional supplementary insight.

Supplementary Insight 5: Categories of accident scenarios that include the more likely LTSBO scenario dominate average individual latent cancer fatality risk within ten miles of each representative NPP site.

Evaluation of the data in Table II-II and Table III-II reveals that, for each representative NPP site, the conditional consequence results varied by an order of magnitude or less across all accident scenarios, while the frequency results varied by several orders of magnitude. For the representative BWR site (Table II-II), conditional average individual latent cancer fatality risk within ten miles ranged from 7.3E-05 (about 1 chance in 10,000 of dying from cancer caused by acute or chronic radiation exposures due to the accident scenario) to 2.8E-04 (about 1 chance in 4,000). By contrast, the scenario frequencies ranged from 2.3E-05 (expected occurrence of about once every 40,000 years) to 3.8E-08 (expected occurrence of about once every 30 million years), with most of the variation occurring within the two-unit accident scenario frequencies. For the representative PWR site (Table III-
II), conditional average individual latent cancer fatality risk within ten miles ranged from 4.7E-05 (about 1 chance in 20,000) to 5.6E-04 (about 1 chance in 2,000). By contrast, the scenario frequencies ranged from 6.8E-05 (expected occurrence of about once every 10,000 years) to 2.7E-11 (expected occurrence of about once every 40 billion years), with most of the variation occurring within the two-unit accident scenario frequencies.

Since scenario-specific risk is calculated as the product of scenario frequency and conditional consequences, the relatively little variation in conditional consequences and large variation in frequencies across all accident scenarios means that the frequency element has the greatest influence on the relative contribution of each scenario to total mean risk. Since categories of accident scenarios that include the more likely LTSBO scenario have higher frequencies, these scenarios dominate average individual latent cancer fatality risk within ten miles of each representative NPP site.

**Insights from Site-Specific Risk Profiles (Appendix IV)**

Figure IV-2 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for average individual latent cancer fatality risk within ten miles of each representative NPP site for each safety goal policy alternative. The results presented in Figure IV-2 further reinforce the finding that categories of accident scenarios that include the more likely LTSBO scenario dominate average individual latent cancer fatality risk within ten miles of both sites.
Insights from Site-Specific Risk Curves (Appendix V)

Figure V-2 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified levels of average individual latent cancer fatality risk within ten miles of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative.

The risk level of interest for average individual latent cancer fatality risk within ten miles of each representative NPP site is 1E-06, which means that individuals residing within ten miles of each site have at least a 1 in 1,000,000 chance of dying from cancer caused by acute or chronic radiation exposures if an accidental radiological release from the site occurs. Like average individual early fatality risk, this risk level is selected because it is commonly used for expressing individual risks and is more than three orders of magnitude below the background risk of all-cause cancer death that individuals in the US are generally exposed to in a given year, based on 2013 data.72

For the representative BWR site (Panel I), the mean frequency of exceeding an average individual latent cancer fatality risk level of 1E-06 is 3E-05 per year for both Option 1 and Option 2. This means that an accidental release from the representative BWR site that results in an average individual latent cancer fatality risk level of greater than 1E-06 within ten miles is expected to occur about once every 30,000 years, regardless of whether only single-unit accident scenarios are considered or both single-unit and two-unit accident scenarios are considered. Although this suggests that including the risk contribution from two-unit accident scenarios has a negligible effect on average individual latent cancer fatality risk
within ten miles of the representative BWR site, Panel I of Figure V-2 shows that including two-unit accident scenarios causes: (1) the mean exceedance frequency to increase for risk levels at or above 1E-04 (1 in 10,000 chance); and (2) the maximum risk level observed across all weather trials and accident scenarios to increase from 5E-04 (1 in 2,000 chance) to 1E-03 (1 in 1,000 chance).

For the representative PWR site (Panel II), the mean frequency of exceeding an average individual latent cancer fatality risk level of 1E-06 is 8E-05 per year for Option 1 and 9E-05 per year for Option 2. This means that an accidental release from the representative PWR site that results in an average individual latent cancer fatality risk level of greater than 1E-06 within ten miles is expected to occur about once every 13,000 years if only single-unit accident scenarios are considered, and about once every 11,000 years if both single-unit and two-unit accident scenarios are considered.

4.C.3. Societal Radiological Health Risk Perspective

This section describes additional insights derived from a more detailed evaluation of the base case analysis results for the societal radiological health risk perspective.

4.C.3.I. Total Early Fatality Risk (0-50 miles)

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-III in Appendix II and Table III-III in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted
risk values for total early fatality risk within 50 miles of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results reveals that—although the absolute values of the conditional consequence and risk results differ—the additional findings pertaining to average individual early fatality risk within one mile described in section 4.C.2.I. also apply to total early fatality risk within 50 miles. More specifically:

1. Categories of accident scenarios involving significant releases of I-131 also dominate total early fatality risk within 50 miles of each representative NPP site;

2. Frequency-conditional consequence tradeoffs for single-unit versus two-unit accident scenarios also differ between the representative NPP sites with respect to total early fatality risk within 50 miles.

3. Relying solely on risk insights for single-unit accident scenarios can also lead to flawed risk management strategies for reducing total early fatality risk within 50 miles.

**Insights from Site-Specific Risk Profiles (Appendix IV)**

Figure IV-3 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for total early fatality risk within 50 miles of each representative NPP site for each safety goal policy alternative. Comparison of Figure IV-3 with Figure IV-1 shows that—although the absolute values of the scenario-specific risk results differ—the site-specific risk profiles for total early fatality risk within 50 miles are nearly identical to those for average individual early fatality risk within one mile.
This means the relative contributions of different categories of accident scenarios are similar for both individual and societal risk measures for early fatalities, and further reinforces the finding that categories of accident scenarios involving significant releases of I-131 dominate early fatality risk for both sites.

**Insights from Site-Specific Risk Curves (Appendix V)**

Figure V-3 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified total numbers of early fatality cases within 50 miles of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative.

The consequence level of interest for total early fatality cases is 1E+00, which means that at least one individual residing within 50 miles of each representative NPP site is expected to die from acute radiation exposure if an accidental radiological release from the site occurs. This consequence level is selected because it is intuitive and represents the level at which any early fatalities are predicted to occur because of accident scenarios involving each representative NPP site.

For the representative BWR site (Panel I), the maximum total number of early fatality cases observed over all weather trials and all modeled accident scenarios is 2E-03 for Option 1 and 2E-02 for Option 2. For the representative PWR site (Panel II), the maximum total number of early fatality cases observed over all weather trials and all modeled accident scenarios is 2E-01 for Option 1 and 5E-01 for Option 2.
Although it seems counterintuitive to have a non-integer result for the total number of early fatality cases that is less than one, this result occurs because of two features of the MACCS code that affect modeling of the population surrounding a NPP site: (1) the assignment of populations to discrete spatial grid elements can cause a fraction of an individual to be assigned to a specific grid element; and (2) population fractions are assigned to each of the six emergency response cohorts defined in section 3.C.4.II, which means a non-integer number can be assigned to each cohort. For these reasons, it is possible for an accident scenario to result in more than zero but less than one early fatality.

To obtain some practical insights from Figure V-3, the mean exceedance frequency for the maximum total number of early fatality cases observed for Option 2 can be used as an upper limit on the mean frequency of accident scenarios that result in at least one early fatality for each NPP site. For the representative BWR site (Panel I), this mean exceedance frequency is 4E-12, which means that an accidental release that results in at least one early fatality among individuals residing within 50 miles is expected to occur no more than about once every 250 billion years, even if both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), this mean exceedance frequency is 2E-14, which means that an accidental release that results in at least one early fatality among individuals residing within 50 miles is expected to occur no more than about once every 50 trillion years, even if both single-unit and two-unit accident scenarios are considered.

As previously stated, a very large asteroid capable of causing a global catastrophe is expected to occur about once every 50 million years, with a 90% confidence interval
that ranges from a frequency of once every 20 million years to one every 1 billion years. These results suggest that—based on the assumptions made in the base case analysis for this dissertation research—the magnitude of total early fatality risk within 50 miles of each representative NPP site is extremely low.

4.C.3.II. Total Latent Cancer Fatality Risk (0-50 miles)

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-IV in Appendix II and Table III-IV in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted risk values for total latent cancer fatality risk within 50 miles of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results reveals that—although the absolute values of the conditional consequence and risk results differ—the additional findings pertaining to average individual latent cancer fatality risk within ten miles described in section 4.C.2.II. also apply to total latent cancer fatality risk within 50 miles. More specifically, the relatively little variation in conditional consequences and large variation in frequencies across all accident scenarios means that the frequency element has the greatest influence on the relative contribution of each scenario to total mean risk. Since categories of accident scenarios that include the more likely LTSBO scenario have higher frequencies, these scenarios also dominate total latent cancer fatality risk within 50 miles of each representative NPP site.
Insights from Site-Specific Risk Profiles (Appendix IV)

Figure IV-4 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for total latent cancer fatality risk within 50 miles of each representative NPP site. Comparison of Figure IV-4 with Figure IV-2 shows that—although the absolute values of the scenario-specific risk results differ—the site-specific risk profiles for total latent cancer fatality risk within 50 miles are nearly identical to those for average individual latent cancer fatality risk within ten miles. This means the relative contributions of different categories of accident scenarios are similar for both individual and societal risk measures for latent cancer fatalities, and further reinforces the finding that categories of accident scenarios that include the more likely LTSBO scenario dominate latent cancer fatality risk for both sites.

Insights from Site-Specific Risk Curves (Appendix V)

Figure V-4 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified total numbers of latent cancer fatality cases within 50 miles of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative.

Like total early fatality cases, the consequence level of interest for total latent cancer fatality cases is 1E00, which means that at least one individual residing within 50 miles of each representative NPP site is expected to die from acute or chronic radiation exposures if an accidental radiological release from the site occurs. This consequence level is selected because it is intuitive and represents the level at which any excess latent cancer fatalities are predicted to occur because of accident
scenarios involving each representative NPP site.

For the representative BWR site (Panel I), the mean frequency of exceeding one latent cancer fatality is $3 \times 10^{-5}$ per year for both Option 1 and Option 2. This means that an accidental release from the representative BWR site that results in at least one excess latent cancer fatality among individuals residing within 50 miles is expected to occur about once every 30,000 years, regardless of whether only single-unit accident scenarios or both single-unit and two-unit accident scenarios are considered.

For the representative PWR site (Panel II), the mean frequency of exceeding one latent cancer fatality is $8 \times 10^{-5}$ per year for Option 1 and $9 \times 10^{-5}$ per year for Option 2. This means that an accidental release from the representative PWR site that results in at least one excess latent cancer fatality among individuals residing within 50 miles is expected to occur about once every 13,000 years if only single-unit accident scenarios are considered, and about once every 11,000 years if both single-unit and two-unit accident scenarios are considered.

4.C.4. Societal Non-Radiological Health Risk Perspective

This section describes additional insights derived from a more detailed evaluation of the base case analysis results for the societal non-radiological health risk perspective.

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-V in Appendix II and Table III-V in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted risk values for total emergency phase population relocation risk within 50 miles of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results reveals that—although the nature and absolute values of the conditional consequence and risk results differ—the additional findings pertaining to latent cancer fatality risk described in section 4.C.2.II. and section 4.C.3.II. also apply to total emergency phase population relocation risk within 50 miles. More specifically, the relatively little variation in conditional consequences and large variation in frequencies across all accident scenarios means that the frequency element has the greatest influence on the relative contribution of each scenario to total mean risk. Since categories of accident scenarios that include the more likely LTSBO scenario have higher frequencies, these scenarios also dominate total emergency phase population relocation risk within 50 miles of each representative NPP site.
**Insights from Site-Specific Risk Profiles (Appendix IV)**

Figure IV-5 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for total emergency phase relocation risk within 50 miles of each representative NPP site. Comparison of Figure IV-5 with Figure IV-2 and Figure IV-4 shows that—although the nature and absolute values of the scenario-specific risk results differ—the site-specific risk profiles for total emergency phase population relocation risk within 50 miles are nearly identical to those for latent cancer fatality risk. This means the relative contributions of different categories of accident scenarios are similar for these different risk measures and further reinforces the finding that categories of accident scenarios that include the more likely LTSBO scenario dominate total emergency phase population relocation risk within 50 miles of both sites.

**Insights from Site-Specific Risk Curves (Appendix V)**

Figure V-5 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified levels of the total population relocated during the emergency phase within 50 miles of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative.

The consequence level of interest for the total population relocated during the emergency phase is 7E04, which means that at least 70,000 individuals residing within 50 miles of each representative NPP site are expected to be relocated during the emergency phase of response to an accidental radiological release from the site. This consequence level is selected because it: (1) corresponds to the approximate
number of individuals relocated during the emergency phase of response to the 2011 Fukushima nuclear accident\(^8\) and (2) can provide insight into the frequency of nuclear accidents that are at least as severe as Fukushima with respect to emergency phase relocation.

For the representative BWR site (Panel I), the mean frequency of exceeding 70,000 relocated individuals is 1E-05 per year for Option 1 and 2E-05 per year for Option 2. This means that an accidental release from the representative BWR site that results in emergency phase relocation of more than 70,000 individuals within 50 miles is expected to occur once every 100,000 years if only single-unit accident scenarios are considered, and once every 50,000 years if both single-unit and two-unit accident scenarios are considered.

For the representative PWR site (Panel II), the mean frequency of exceeding 70,000 relocated individuals is 5E-05 per year for Option 1 and 6E-05 per year for Option 2. This means that an accidental release from the representative PWR site that results in emergency phase relocation of more than 70,000 individuals within 50 miles is expected to occur once every 20,000 years if only single-unit accident scenarios are considered, and once every 17,000 years if both single-unit and two-unit accident scenarios are considered.
4.C.4.II. Total Recovery Phase Population Relocation Risk (0-50 miles)

Insights from Scenario-Specific Frequency and Conditional Consequence Results for Each Representative NPP Site (Appendix II and Appendix III)

For each single-unit and two-unit accident scenario modeled as part of this dissertation research, Table II-VI in Appendix II and Table III-VI in Appendix III illustrate the mean frequency, mean conditional consequence, and mean adjusted risk values for total recovery phase population relocation risk within 50 miles of the representative BWR and PWR sites, respectively. Evaluation of these detailed scenario-specific results yields two additional supplementary insights—one that is unique to each representative NPP site.

**Supplementary Insight 6:** Total recovery phase population relocation risk within 50 miles of the representative BWR site is dominated by categories of accident scenarios that: (1) are more likely to occur; or (2) progress to radiological release more rapidly and release more Cesium-137 (Cs-137).

Evaluation of the data for the representative BWR site (Table II-VI) reveals that the conditional consequence results varied by about an order of magnitude across all accident scenarios, while the frequency results varied by several orders of magnitude. The expected total number of individuals residing within 50 miles of the representative BWR site relocated during the recovery phase of response ranged from 30,000 to 420,000. By contrast, the scenario frequencies ranged from 2.3E-05 per year (expected occurrence of about once every 40,000 years) to 3.8E-08 per year (expected occurrence of about once every 30 million years). Since scenario-specific risk is calculated as the product of scenario frequency and conditional consequences,
the relatively little variation in conditional consequences and large variation in
frequencies across all accident scenarios should mean that the frequency element
has the greatest influence on the relative contribution of each scenario to total mean
risk. Since categories of accident scenarios that include the more likely LTSBO
scenario have higher frequencies, these scenarios should dominate total recovery
phase population relocation risk within 50 miles of the representative BWR site.

However, Table II-VI shows that two single-unit accident scenarios comprise 72% of
the total mean value for total recovery phase relocation risk: (1) LTSBO (30%); and
(2) STSBO-Base (42%). Further evaluation of the data reveals that the single-unit
STSBO-Base scenario results in the maximum expected total number of individuals
residing within 50 miles of the representative BWR that are relocated during the
recovery phase (420,000). Combining this accident scenario with others to create
two-unit accident scenarios does not result in a greater number of relocated
individuals, which suggests the STSBO-Base scenario alone is sufficient to reach a
plateau or limit on the expected number of relocated individuals.

One plausible explanation for this observation is that—as shown in Table III of
Chapter 1—the STSBO-Base scenario has the shortest amount of time from accident
initiation to radiological release (8 hours), indicating it is a relatively rapidly
progressing accident scenario. If more quantities of radiological materials are
released during a shorter time-period, this reduces the effect of changing weather
conditions (especially wind direction) on dispersion of the radiological materials.
Thus, areas of land near the NPP site can be contaminated with greater levels of
radioactivity, thereby increasing the total number of individuals relocated during
the recovery phase.
Another plausible explanation for this observation is that—as shown in Table III of Chapter 1—the STSBO-Base scenario results in the largest Cs-137 release fraction (2% of reactor core inventory) among the single-unit accident scenarios modeled for the representative BWR site. Due to its long half-life\(^{\text{vv}}\) of about 30 years, Cs-137 persists in the environment for long time-periods and has been shown in previous studies\(^{19-21}\) to be an important contributor to land contamination risk, which influences total recovery phase population relocation risk, since the extent of land contamination coupled with the habitability dose criterion determine whether individuals are relocated during the recovery phase.

Regardless of explanation, while the category of accident scenarios represented by the STSBO-Base scenario is an order of magnitude less likely to occur than the category represented by the LTSBO scenario (2.3E-06 per year versus 2.3E-05 per year), the STSBO-Base scenario results in conditional consequences that are more than an order of magnitude greater than the LTSBO scenario (420,000 relocated individuals versus 30,000 relocated individuals). Together, these factors make the category of accident scenarios represented by the STSBO-Base scenario a more important contributor to total recovery phase population relocation risk within 50 miles of the representative BWR site.

\(^{\text{vv}}\) The half-life for a radionuclide represents the amount of time required for radioactive decay processes to reduce the quantity of a radionuclide present at a specified time to one-half of that quantity.
**Supplementary Insight 7: Total recovery phase population relocation risk within 50 miles of the representative PWR site is dominated by less likely categories of accident scenarios that: (1) progress to radiological release more rapidly; and/or (2) release more Cs-137.**

Evaluation of the data for the representative PWR site (Table III-VI) reveals a slightly different pattern than what was observed for the representative BWR site. For the representative PWR site, both the frequency and conditional consequence results varied by several orders of magnitude across all accident scenarios. The expected total number of individuals residing within 50 miles of the representative PWR site relocated during the recovery phase of response ranged from 1 to 48,000, with most of the variation occurring within the conditional consequence results for single-unit accident scenarios. By contrast, the scenario frequencies ranged from 6.8E-05 (expected occurrence of about once every 10,000 years) to 2.7E-11 (expected occurrence of about once every 40 billion years), with most of the variation occurring within the two-unit accident scenario frequencies.

Since single-unit accident scenarios dominate total recovery phase population relocation risk within 50 miles of the representative PWR site, comprising 71% of the total mean value, the variation in conditional consequence results among single-unit accident scenarios appears to have the greatest influence on the relative contribution of each scenario to total mean risk. Table III-VI shows that two single-unit accident scenarios comprise this 71% of the total mean value for total recovery phase relocation risk: (1) STSBO-TISGTR (52%); and (2) ISLOCA (19%). The single-unit STSBO-TISGTR scenario is expected to result in relocation of 6.4E+03 (6,400) individuals during the recovery phase, with a mean frequency of 1.4E-06 per year.
The single-unit ISLOCA scenario is expected to result in relocation of 3.0E+04 (30,000) individuals during the recovery phase, with a mean frequency of 1.0E-07 per year (expected occurrence of once every 10 million years). Further evaluation of the data in Table III-VI reveals two findings that provide additional insight into why these two single-unit accident scenarios dominate total recovery phase population relocation risk within 50 miles of the representative PWR site: (1) the other two single-unit accident scenarios (LTSBO and STSBO-Base)—although more likely to occur—are expected to result in a relatively small number of relocated individuals (1 and 10 individuals, respectively); and (2) the conditional consequence results for these two single-unit accident scenarios are within an order of magnitude of the maximum conditional consequence results observed for the far less likely two-unit accident scenarios involving each of these accident scenarios.

Table III of Chapter 1 identifies characteristics of the two single-unit accident scenarios that dominate total recovery phase population relocation risk for the representative PWR site. As previously discussed, the STSBO-TISGTR scenario is the most rapidly progressing accident scenario, with only three hours from accident initiation to the onset of core damage, and only 0.5 hours from onset of core damage to release of radiological materials to the environment. The ISLOCA scenario is the next most rapidly progressing accident scenario, with only 13 hours from accident initiation to the onset of core damage, and radiological release occurring instantaneously upon core damage. In addition, the ISLOCA scenario results in the largest Cs-137 release fraction (2% of reactor core inventory) among the single-unit accident scenarios modeled for the representative PWR site. As previously described,
the long-lived Cs-137 radionuclide has been shown in previous studies\textsuperscript{19-21} to be an important contributor to land contamination risk, which influences total recovery phase population relocation risk, since the extent of land contamination coupled with the habitability dose criterion determine whether individuals are relocated during the recovery phase.

Combining these characteristics, one plausible explanation for why these two rapidly progressing single-unit accident scenarios dominate total recovery phase population relocation risk is like an explanation that was provided for the representative BWR site. If more quantities of radiological materials are released during a shorter time-period, this reduces the effect of changing weather conditions (especially wind direction) on dispersion of the radiological materials. Thus, areas of land near the NPP site can be contaminated with greater levels of radioactivity, thereby increasing the total number of individuals relocated during the recovery phase.

**Insights from Site-Specific Risk Profiles (Appendix IV)**

Figure IV-6 in Appendix IV presents the site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for total recovery phase population relocation risk within 50 miles of each representative NPP site. The results presented in Panel II further reinforce the finding that total recovery phase population relocation risk within 50 miles of the representative BWR site is dominated by categories of accident scenarios that: (1) include the more likely LTSBO scenario; or (2) include the rapidly progressing STSBO-Base scenario that also releases the most Cs-137. The results presented in
Panel IV further reinforce the finding that total recovery phase population relocation risk within 50 miles of the representative PWR site is dominated by categories of accident scenarios that include: (1) the rapidly progressing STSBO-TISGTR scenario; or (2) the rapidly progressing ISLOCA scenario that also releases the most Cs-137.

**Insights from Site-Specific Risk Curves (Appendix V)**

Figure V-6 in Appendix V presents site-specific risk curves that illustrate the mean frequency of exceeding specified levels of the total population relocated during the recovery phase within 50 miles of each representative NPP site, considering the full spectrum of accident scenarios modeled for each safety goal policy alternative.

The consequence level of interest for the total population relocated during the recovery phase is 1E05, which means that at least 100,000 individuals residing within 50 miles of each representative NPP site are expected to be relocated during the recovery phase of response to an accidental radiological release from the site. Like the total population relocated during the emergency phase, this consequence level is selected because it: (1) corresponds to the approximate number of individuals relocated during the recovery phase of response to the 2011 Fukushima nuclear accident; and (2) can provide insight into the frequency of nuclear accidents that are at least as severe as Fukushima with respect to recovery phase relocation.

For the representative BWR site (Panel I), the mean frequency of exceeding 100,000 relocated individuals is 4E-06 per year for Option 1 and 5E-06 per year for Option 2. This means that an accidental release from the representative BWR site that results in recovery phase relocation of more than 100,000 individuals within 50 miles is
expected to occur once every 250,000 years if only single-unit accident scenarios are considered, and once every 200,000 years if both single-unit and two-unit accident scenarios are considered.

For the representative PWR site (Panel II), the mean frequency of exceeding 100,000 relocated individuals is 6E-09 per year for Option 1 and 8E-09 per year for Option 2. This means that an accidental release from the representative PWR site that results in recovery phase relocation of more than 100,000 individuals within 50 miles is expected to occur about once every 170 million years if only single-unit accident scenarios are considered, and about once every 130 million years if both single-unit and two-unit accident scenarios are considered.
4.D. One-Way Sensitivity Analysis: Effect of Variation in Assumed Level of Dependence Between Co-Located Reactor Units (Aim 2)

The base case analysis for each representative NPP site relied on two assumptions that respectively influence the frequency and conditional consequence elements of the risk triplet for two-unit accident scenarios: (1) the level of dependence between co-located reactor units was assumed to be 0.1, which implies a 10% chance of a co-located unit experiencing a concurrent accident scenario, given that a single-unit accident scenario involving any unit occurs at the site; and (2) the timing offset (delay time) between concurrent accident scenarios involving co-located units that comprise each two-unit accident scenario was assumed to be zero, which implies the concurrent accident scenarios occur simultaneously.

This section presents and interprets results for one-way sensitivity analyses that were performed for both representative NPP sites to evaluate the effect of variation in the assumed level of dependence between co-located reactor units on: (1) the FOM results for all selected risk metrics; and (2) the QHO margin results for the early fatality risk and latent cancer fatality risk QHOs. For these sensitivity analyses, the global average parameter used to model the level of inter-unit dependence across all two-unit accident scenarios ($\beta$) was varied over its full range of possible values, from 0 (no dependence between co-located reactor units, which is equivalent to assuming the units are completely independent) to 1 (complete dependence between co-located reactor units).
4.D.1. Figure of Merit Results

One-way sensitivity analyses performed to evaluate the effect of variation in the assumed level of dependence between co-located reactor units resulted in one additional key finding with respect to the FOM results.

Key Finding 3: The relative contribution of two-unit accident scenarios to the total mean value for all selected risk metrics increases as the assumed level of inter-unit dependence increases, with two-unit accident scenarios becoming the dominant risk contributors at higher assumed levels of inter-unit dependence.

Table IX illustrates summary results for the one-way sensitivity analysis by risk metric for the representative BWR site. As shown, the FOM ranges from 23% to 64% across all selected risk metrics for the base case, and ranges from 74% to 95% for the worst sensitivity case of $\beta = 1$, which assumes complete dependence between co-located reactor units. Although the relative importance of two-unit accident scenarios varies by risk metric—with two-unit accident scenarios being more important to risk metrics related to early fatality risk—the relative contribution of two-unit accident scenarios to total mean risk is more than 50% across all risk metrics. Two-unit accident scenarios are thus considered the dominant contributor to total mean risk for the representative BWR site when the two co-located reactor units are assumed to be completely dependent.

Table X illustrates summary results for the one-way sensitivity analysis by risk metric for the representative PWR site. As shown, the FOM ranges from 17% to 29% across all selected risk metrics for the base case, and ranges from 67% to 80% for the
worst sensitivity case of $\beta = 1$, which assumes complete dependence between co-located reactor units. Although the relative importance of two-unit accident scenarios still varies to some extent by risk metric, the relative contribution of two-unit accident scenarios to total mean risk is more than 50% across all risk metrics. Thus, two-unit accident scenarios are also considered the dominant contributor to total mean risk for the representative PWR site when the two co-located reactor units are assumed to be completely dependent.

Figure 12 further displays the effect of variation in the assumed level of inter-unit dependence on the FOM results for all selected risk metrics for each representative NPP site, as the level of inter-unit dependence is varied over its full range of possible values from 0 to 1. 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 accident scenarios compared to single-unit accident scenarios. The effect of including the contribution from two-unit accident scenarios should thus increase as $\beta$ increases, and this is what is observed.

The non-linear increase in the FOM as $\beta$ increases is because the total mean risk under Option 2 is calculated as the sum of the risk contributions from single-unit and two-unit accident scenarios. Since $\beta$ only influences the two-unit accident scenario contribution, increasing $\beta$ results in a non-linear increase in the relative contribution of two-unit accident scenarios. Figure 12 also shows that, for both representative NPP sites, the relative contribution from two-accident scenarios is more than 10%—and is thus considered non-negligible—for all selected risk metrics at assumed levels of inter-unit dependence that exceed 5% ($\beta > 0.05$).
For the representative BWR site (Panel I), the risk metrics related to early fatality risk (average individual early fatality risk within one mile and total early fatality risk within 50 miles) are more sensitive to the assumed level of inter-unit dependence in the region of $0 \leq \beta \leq 0.1$ and less sensitive in the region of $0.1 \leq \beta \leq 1$. However, the most important finding from a risk management perspective is that for all risk metrics, two-unit accident scenarios become the dominant contributor to total mean risk for assumed levels of inter-unit dependence greater than 35% ($\beta > 0.35$). This suggests that for BWR sites that are judged to have a level of inter-unit dependence greater than 35%, including only the risk contribution from single-unit accident scenarios would: (1) capture less than half of the total accident risk; and (2) result in biased inferences and risk management strategies.

For the representative PWR site (Panel II), total emergency phase population relocation risk within 50 miles is slightly less sensitive to the assumed level of inter-unit dependence in the region of $0 \leq \beta \leq 0.5$ and slightly more sensitive in the region of $0.5 \leq \beta \leq 1$. However, the most important finding from a risk management perspective is that for all other risk metrics, two-unit accident scenarios become the dominant contributor to total mean risk for assumed levels of inter-unit dependence greater than 25% ($\beta > 0.25$). This suggests that for PWR sites that are judged to have a level of inter-unit dependence greater than 25%, including only the risk contribution from single-unit accident scenarios would: (1) capture less than half of the total accident risk; and (2) result in biased inferences and risk management strategies.
Table IX. Inter-Unit Dependence Sensitivity Analysis Summary Results for Representative BWR Site (Peach Bottom)

<table>
<thead>
<tr>
<th>Risk Metric</th>
<th>Figure of Merit</th>
<th>Option 2 QHO Margin&lt;sup&gt;b&lt;/sup&gt;</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Base Case β = 0.1&lt;sup&gt;a&lt;/sup&gt;</td>
<td>Worst Case β = 1</td>
</tr>
<tr>
<td>Average Individual Early Fatality Risk (0-1 mile)</td>
<td>61%</td>
<td>94%</td>
</tr>
<tr>
<td>Average Individual Latent Cancer Fatality Risk (0-10 miles)</td>
<td>23%</td>
<td>75%</td>
</tr>
<tr>
<td>Total Early Fatality Risk (0-50 miles)</td>
<td>64%</td>
<td>95%</td>
</tr>
<tr>
<td>Total Latent Cancer Fatality Risk (0-50 miles)</td>
<td>26%</td>
<td>78%</td>
</tr>
<tr>
<td>Total Emergency Phase Population Relocation Risk (0-50 miles)</td>
<td>24%</td>
<td>75%</td>
</tr>
<tr>
<td>Total Recovery Phase Population Relocation Risk (0-50 miles)</td>
<td>25%</td>
<td>74%</td>
</tr>
</tbody>
</table>

<sup>a</sup> For β = 0, which is equivalent to assuming there are no inter-unit dependencies that could give rise to two-unit accident scenarios given the occurrence of a single-unit accident scenario involving another unit at the site, including the risk contribution from two-unit accident scenarios has no effect (i.e., FOM = 0%). These trivial results for the β = 0 sensitivity case are not displayed.

<sup>b</sup> Since the level of inter-unit dependence does not influence the risk results for single-unit accident scenarios, the Option 1 QHO margin results are not influenced by variation in the assumed level of inter-unit dependence, and therefore are not displayed.
Table X. Inter-Unit Dependence Sensitivity Analysis Summary Results for Representative PWR Site (Surry)

<table>
<thead>
<tr>
<th>Risk Metric</th>
<th>Figure of Merit</th>
<th>Option 2 QHO Margin</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Base Case β = 0.1&lt;sup&gt;a&lt;/sup&gt;</td>
<td>Worst Case β = 1</td>
</tr>
<tr>
<td>Average Individual Early Fatality Risk (0-1 mile)</td>
<td>28%</td>
<td>80%</td>
</tr>
<tr>
<td>Average Individual Latent Cancer Fatality Risk (0-10 miles)</td>
<td>27%</td>
<td>79%</td>
</tr>
<tr>
<td>Total Early Fatality Risk (0-50 miles)</td>
<td>29%</td>
<td>80%</td>
</tr>
<tr>
<td>Total Latent Cancer Fatality Risk (0-50 miles)</td>
<td>27%</td>
<td>79%</td>
</tr>
<tr>
<td>Total Emergency Phase Population Relocation Risk (0-50 miles)</td>
<td>17%</td>
<td>67%</td>
</tr>
<tr>
<td>Total Recovery Phase Population Relocation Risk (0-50 miles)</td>
<td>29%</td>
<td>79%</td>
</tr>
</tbody>
</table>

<sup>a</sup> For β = 0, which is equivalent to assuming there are no inter-unit dependencies that could give rise to two-unit accident scenarios given the occurrence of a single-unit accident scenario involving another unit at the site, including the risk contribution from two-unit accident scenarios has no effect (i.e., FOM = 0%). These trivial results for the β = 0 sensitivity case are not displayed.

<sup>b</sup> Since the level of inter-unit dependence does not influence the risk results for single-unit accident scenarios, the Option 1 QHO margin results are not influenced by variation in the assumed level of inter-unit dependence, and therefore are not displayed.
Figure 12. Effect of Variation in Assumed Level of Inter-Unit Dependence on Figure of Merit Results for Each Representative NPP Site. Increasing the assumed level of inter-unit dependence ($\beta$) effectively increases the frequency element of the risk contribution from two-unit accident scenarios, thereby increasing the relative importance of two-unit accident scenarios compared to single-unit accident scenarios. The relative contribution of two-unit accident scenarios to the total mean value for all selected risk metrics thus increases as $\beta$ is varied over its full range from 0 (no dependence between co-located reactor units) to 1 (complete dependence between co-located reactor units). Panel I shows that risk metrics related to early fatality risk for the representative BWR site are more sensitive to variation in the assumed level of inter-unit dependence for $0 \leq \beta \leq 0.1$ than other selected risk metrics. Panel II shows that the total emergency phase population relocation risk metric is less sensitive to variation in the assumed level of inter-unit dependence for $0 \leq \beta \leq 0.5$. 
4.D.2. Quantitative Health Objective Results

One-way sensitivity analyses to evaluate the effect of variation in the assumed level of dependence between co-located reactor units resulted in one additional key finding with respect to the safety goal QHO results.

**Key Finding 4:** Although increasing the assumed level of inter-unit dependence increases the relative contribution of two-unit accident scenarios to total mean risk, sufficient margin to each QHO remains even under the worst-case assumption of complete inter-unit dependence.

Table IX and Table X illustrate summary results for the effect of variation in the assumed level of inter-unit dependence on the margin to both the early fatality risk and latent cancer fatality risk QHOs for the representative BWR site and representative PWR site, respectively. As described in section 4.D.1., increasing the assumed level of inter-unit dependence increases the relative contribution of two-unit accident scenarios to the total mean value for all selected risk metrics—including those that are compared against the QHOs for early fatality risk and latent cancer fatality risk to measure attainment of the safety goals (average individual early fatality risk within one mile and average individual latent cancer fatality risk within ten miles, respectively). Table IX and Table X illustrate that, while the total mean value of these risk metrics under Option 2 increases as the assumed level of inter-unit dependence and relative contribution of two-unit accident scenarios increases, the corresponding reduction in margin to each QHO is less than an order of magnitude for each representative NPP site, even for the worst-case assumption of complete inter-unit dependence. Moreover, even under this
worst-case assumption, the minimum margin to a QHO is $9.2\times 10^1$—which is the margin to the latent cancer fatality risk QHO for the representative PWR site. This means that even if the two co-located reactor units at each representative NPP site were completely dependent (i.e., an accident scenario involving any unit at the site would always result in a concurrent accident scenario involving the co-located unit), the average individual latent cancer fatality risk within ten miles of the site would have to increase by two or more orders of magnitude for the corresponding QHO and safety goal to no longer be satisfied. The margin to the early fatality risk QHO is even greater for each site, requiring an increase in average individual early fatality risk within one mile of three or more orders of magnitude for the QHO and corresponding safety goal to no longer be satisfied.

Figure 13 displays the effect of variation in the assumed level of inter-unit dependence on the margin to both the early fatality risk and latent cancer fatality risk QHOs for the representative BWR site (Panel I and Panel II, respectively) and the representative PWR site (Panel III and Panel IV, respectively). As shown, the margins to each QHO for Option 1 and Option 2 are equal when $\beta = 0$. This is expected, since an assumption of no inter-unit dependence means that single-unit accident scenarios in any unit on the site can never give rise to two-unit accident scenarios, and thus including the risk contribution from two-unit accident scenarios has no effect. Figure 13 also shows that variation in the assumed level of inter-unit dependence has no effect on the Option 1 QHO margin, which is also expected since only single-unit accident scenarios are considered, and inter-unit dependence does not influence the risk contribution from single-unit accident scenarios. By contrast, as the assumed level of inter-unit dependence increases beyond zero, the total mean
risk under Option 2 for each risk metric increases non-linearly as shown in Figure 12, and the corresponding margin to each QHO thus decreases non-linearly as shown in Figure 13.

Consistent with the base case analysis, the related policy implication is that—assuming the QHOs continue to be limited to existing measures of individual risk of experiencing radiological health effects—expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios would likely not impact safety-related decisions for safety goal policy applications. This is because—from an individual radiological health risk perspective—these results indicate the residual risk is already at an acceptably low level, even under worse-case assumptions of complete inter-unit dependence. Thus, limited resources that would be applied to a proposed regulatory action could be better applied to alternative courses of action.
Figure 13. Effect of Variation in Assumed Level of Inter-Unit Dependence on Margin to Each Quantitative Health Objective for Each Representative NPP Site and Each Safety Goal Policy Alternative. As the assumed level of inter-unit dependence increases beyond zero, the total mean risk under Option 2 for each safety goal QHO risk metric increases non-linearly as shown in Figure 12, and the corresponding margin to each QHO thus decreases non-linearly as shown in this figure. Note that a logarithmic scale has been used for the vertical axis to display order of magnitude changes in the QHO margin.

As previously stated, the base case analysis for each representative NPP site relied on two assumptions that respectively influence the frequency and conditional consequence elements of the risk triplet for two-unit accident scenarios: (1) the level of dependence between co-located reactor units was assumed to be 0.1, which implies a 10% chance of a co-located unit experiencing a concurrent accident scenario, given that a single-unit accident scenario involving any unit occurs at the site; and (2) the timing offset (delay time) between concurrent accident scenarios involving co-located units that comprise each two-unit accident scenario was assumed to be zero, which implies the concurrent accident scenarios occur simultaneously.

This section presents and interprets results for one-way sensitivity analyses that were performed for both representative NPP sites to evaluate the effect on FOM results for all selected risk metrics of variation in the assumed timing offset between concurrent accident scenarios involving co-located units that comprise each two-unit accident scenario. As described in section 3.C.3.I., a timing offset parameter is used in the MACCS multi-source model 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. For these sensitivity analyses, the value of this timing offset parameter was varied from 0 days in the base case to 7 days, in one-day increments. Figure 14 illustrates the effect of variation in this timing offset parameter on the FOM results for all selected risk metrics for each representative NPP site.
Panel I and Panel II show that the FOM results for risk metrics related to early fatality risk are sensitive to variation in the assumed timing offset between concurrent accident scenarios involving co-located reactor units that comprise each two-unit accident scenario in the region of 0 to 1 day. As the timing offset is increased from 0 to 1 day (i.e., a one-day delay is introduced between the accident scenario in the reference unit and the concurrent accident scenario in the co-located unit), the relative contribution of two-unit accident scenarios to risk metrics related to early fatality risk decreases significantly, with relatively little further reduction as the timing offset is increased beyond 1 day. Panel I further shows that the FOM result for average individual latent cancer fatality risk within ten miles of the representative BWR site increases slightly as the timing offset is increased from 0 to 2 days, with relatively little change as the timing offset is increased beyond 2 days.
Key Finding 5: The relative contribution of two-unit accident scenarios to the total mean value for risk metrics related to early fatality risk is sensitive to the assumed timing offset between concurrent accident scenarios.

Panel I and Panel II of Figure 14 show that the FOM results for risk metrics related to early fatality risk are sensitive to variation in the assumed timing offset between concurrent accident scenarios involving co-located reactor units that comprise each two-unit accident scenario in the region of 0 to 1 day. As the timing offset is increased from 0 to 1 day (i.e., a one-day delay is introduced between the accident scenario in the reference unit and the concurrent accident scenario in the co-located unit), the relative contribution of two-unit accident scenarios to risk metrics related to early fatality risk decreases significantly, with relatively little further reduction as the timing offset is increased beyond 1 day.

As described in section 4.C.2.I. and section 4.C.3.I, categories of accident scenarios involving significant releases of I-131 dominate both average individual early fatality risk within one mile and total early fatality risk within 50 miles of each representative NPP site. Two single-unit accident scenarios result in release of more than 10% of the I-131 stored in the reactor core: (1) STSBO-Base, which has an I-131 release fraction of 12% for the representative BWR site; and (2) ISLOCA, which has an I-131 release fraction of 16% for the representative PWR site. In addition to releasing relatively large amounts of I-131, these single-unit accident scenarios are also among the more rapidly progressing accident scenarios. As shown in Table III of Chapter 1, the STSBO-Base scenario has a time to core damage of 1 hour after accident initiation, and a time to radiological release of 8 hours after accident initiation. For the ISLOCA scenario, these times are 13 hours and 13 hours,
respectively; this means that radiological release to the environment occurs
instantaneously upon core damage in the ISLOCA scenario. For both categories of
accident scenarios, there is relatively little time for OROs to implement emergency
response protective actions (e.g., evacuation, sheltering, dose-dependent relocation)
among the affected population.

As previously stated, early fatalities are a type of deterministic radiological health
effect described in section 2.E.1. of Chapter 2. For an early fatality to occur, many
cells must die to cause sufficient damage to tissues. This typically requires high
doses of radiation over short periods of time to exceed dose thresholds for exposed
tissues. Once threshold doses for exposed tissues are exceeded, deterministic health
effects—including early fatalities—are predicted to occur with certainty, with the
severity of the effect increasing as the dose level increases. A plausible explanation
for why the relative contribution of two-unit accident scenarios to risk metrics
related to early fatality risk decreases as the timing offset increases is that
introducing a time delay between concurrent accident scenarios involving these
rapidly progressing scenarios that release large amounts of I-131 can provide
additional time for the offsite weather conditions (especially wind direction) to
change, thereby dispersing the released I-131 over a larger area. This increased
dispersion would thus reduce the concentration of I-131 in areas surrounding the
NPP site and thereby reduce the radiological dose to exposed individuals. For some
fraction of the exposed population, this reduction in the radiological dose could be
sufficient to avoid exceeding the dose threshold for early fatalities, thereby reducing
the contribution to early fatality risk from these two-unit accident scenarios.
Despite this observed sensitivity of risk metrics related to early fatality risk, it is worth noting that for all selected risk metrics, the FOM did not decrease below the 10% criterion as the assumed timing offset was varied. This indicates the relative contribution from two-unit accident scenarios continues to be non-negligible over a range of plausible alternative timing offsets for all selected risk metrics.

**Key Finding 6: Assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.**

For some applications that consider multi-unit accident scenarios (e.g., decisions with respect to siting of multiple reactor units at a shared NPP site), a common practice is to assume that potential concurrent accident scenarios involving co-located units occur simultaneously. Using this assumption, rather than explicitly modeling the radiological releases from multiple units, the analyst instead multiplies the quantities of radionuclides released from a single unit by the number of units being considered. This practice assumes the co-located units are identical and in effect scales a single-unit release to account for the release contributions from co-located units. The underlying hypothesis that supports this common practice is that an assumption of simultaneous releases from multiple units represents a worst-case or bounding scenario.

However, Panel I of Figure 14 shows that the FOM result for average individual latent cancer fatality risk within ten miles of the representative BWR site increases slightly as the timing offset is increased from 0 to 2 days, with relatively little change as the timing offset is increased beyond 2 days. This counter-intuitive
finding suggests that assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.

The SOARCA pilot study\textsuperscript{16-18} showed that latent cancer fatality risk is dominated by long-term exposures that occur during the recovery phase of response to an accident scenario. As with early fatality risk, one plausible explanation for this observation is that increasing the timing offset between concurrent accident scenarios effectively provides more time for offsite weather conditions (especially wind direction) to change between releases of sequential plume segments. Larger areas of land would thus be contaminated at lower levels of radioactivity. For this reason, larger areas of contaminated land would then be able to be returned to habitability for a specified habitability criterion\textsuperscript{ww}. 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\textsuperscript{xx}, this increase in the number of individuals exposed to low levels of radiation would necessarily result in a

\textsuperscript{ww} 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 and thus relocated populations would be allowed to return to the area.

\textsuperscript{xx} 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 dissertation research, only the LNT-based dose-response model from the base case analyses was used; no additional sensitivity analyses were performed to evaluate the effect on results of using plausible alternative dose-response models.
proportional increase in the average individual latent cancer fatality risk.

Using this reasoning, it is possible for this effect to be observed for the representative BWR site and not for the representative PWR site for at least three reasons: (1) the source term characteristics (e.g., timing, energy content, radionuclide composition) for modeled two-unit accident scenarios are different for each site; (2) the offsite population distributions within ten miles of each site are different; and (3) the OROs for each site use different habitability criteria for determining when relocated individuals are allowed to return home to land that has been contaminated, which means that returning individuals could therefore be exposed to different residual levels of contamination for the duration of the long-term phase for each site. Together, this means it is possible for variation in the assumed timing offset between concurrent accident scenarios to have different effects on the FOM results for average individual latent cancer fatality risk within ten miles of each representative NPP site. However, additional research would be needed to formulate and test alternative hypotheses to obtain further insights into the underlying causes for this observed behavior.
4.F. Chapter Summary

This chapter presented and interpreted the results of a policy analysis performed to evaluate the effect of a hypothetical expansion in the scope of the safety goal policy to address two known limitations: (1) exclusion of the risk contribution from multi-unit accident scenarios; and (2) use of risk metrics that are limited to measures of individual risk of experiencing radiological health effects and that do not include measures of the societal risk of radiological and non-radiological health consequences. Several key findings based on both the FOM used to evaluate the effect of including the risk contribution from multi-unit accident scenarios, as well as margins to each of the existing safety goal QHOs, were identified and described:

1. Including the risk contribution from multi-unit accident scenarios results in a non-negligible increase in the total mean value for all selected risk metrics.
2. Including the risk contribution from multi-unit accident scenarios to existing safety goal QHO risk metrics would likely not impact safety-related decisions for safety goal policy applications.
3. The relative contribution of two-unit accident scenarios to the total mean value for all selected risk metrics increases as the assumed level of inter-unit dependence increases, with two-unit accident scenarios becoming the dominant risk contributors at higher assumed levels of inter-unit dependence.
4. Although increasing the assumed level of inter-unit dependence increases the relative contribution of two-unit accident scenarios to total mean risk, sufficient margin to each QHO remains even under the worst-case assumption of complete inter-unit dependence.
5. The relative contribution of two-unit accident scenarios to the total mean value for risk metrics related to early fatality risk is sensitive to the assumed timing offset between concurrent accident scenarios. However, for all selected risk metrics, the FOM did not decrease below 10% as the assumed timing offset was varied, which indicates the relative contribution from two-unit accident scenarios continues to be non-negligible over a range of plausible alternative timing offsets.

6. Assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.

This chapter further described the need for more complete characterizations of risk to determine whether additional risk-based safety goal policy insights can be obtained by expanding the scope of the safety goal policy and QHOs beyond the individual radiological health risk perspective to include the societal radiological and non-radiological health risk perspectives. Using the base case analysis results, additional insights for each of these public health perspectives were developed from three additional ways of characterizing and presenting risk results: (1) scenario-specific frequency and conditional consequence results for each representative NPP site (Appendix II and Appendix III); (2) site-specific risk profiles that illustrate the relative contributions of different categories of accident scenarios to the total mean value for each risk metric (Appendix IV); and (3) site-specific risk curves that illustrate the mean frequency of exceeding specified consequence levels for each risk metric (Appendix V).
From this more complete characterization of risk for each public health perspective, many supplementary insights were identified and described:

1. Categories of accident scenarios involving significant releases of I-131 dominate risk metrics related to early fatality risk for each representative NPP site.

2. Frequency-conditional consequence tradeoffs for single-unit versus two-unit accident scenarios differ between the representative NPP sites with respect to early fatality risk.

3. Relying solely on risk insights for single-unit accident scenarios can lead to flawed risk management strategies for risk metrics that involve threshold effects (e.g., deterministic radiological health effects such as early fatalities).

4. Including the contribution from two-unit accident scenarios shifts risk curves: (1) up, which means that for a specified consequence level, the frequency of exceeding that consequence level increases; and/or (2) to the right, which means the maximum consequence level observed across all accident scenarios and all weather trials increases.

5. Categories of accident scenarios that include the more likely LTSBO scenario dominate a broad range of risk metrics for each representative NPP site, including: (1) average individual latent cancer fatality risk within ten miles; (2) total latent cancer fatality risk within 50 miles; and (3) total emergency phase population relocation risk within 50 miles.

6. Total recovery phase population relocation risk within 50 miles of the representative BWR site is dominated by categories of accident scenarios that: (1) are more likely to occur; or (2) progress to radiological release more rapidly and release more Cs-137.
7. Total recovery phase population relocation risk within 50 miles of the representative PWR site is dominated by less likely categories of accident scenarios that: (1) progress to radiological release more rapidly; and/or (2) release more Cs-137.

These supplementary insights illustrate the potential value of using a broader set of public health risk metrics that go beyond the existing measures of individual risk of experiencing radiological health effects to include measures of the societal risk for radiological and non-radiological health consequences. In addition, they can provide a valuable input to current USNRC and nuclear industry stakeholder deliberations about whether and to what extent the safety goal policy should be expanded to include QHOs that address societal risk.²²,²⁸

Chapter 5 will present the key conclusions and recommendations that stem from this dissertation research.
Chapter 5. Conclusions and Recommendations

5.A. Chapter Introduction and Overview

This chapter presents the key conclusions and recommendations that stem from this dissertation research. It will begin by providing a high-level summary of the research that was performed. Next, key conclusions derived from the findings and insights that were presented in Chapter 4 are summarized. The chapter then identifies the assumptions and limitations in study design that are judged to have the greatest potential for introducing bias or limiting the generalizability of study findings, along with recommendations for future research to address known issues. Finally, the chapter concludes with some additional public health perspectives on improving the quality of probabilistic risk analysis (PRA) for commercial nuclear power plants (NPPs) to enhance its utility as a decision support and communication tool for a diverse set of stakeholder groups.
5.B. High-Level Research Summary

The overall aim for this dissertation research was to evaluate the effect of a hypothetical expansion in the scope of the safety goal policy to include consideration of: (1) the risk contribution from multi-unit accident scenarios for multi-unit NPP sites; and (2) a broader set of public health risk metrics that includes measures of societal risk for radiological and non-radiological health consequences. To do this, two safety goal policy alternatives were identified for evaluation: (1) a status quo option (Option 1) that reflects the existing application of the safety goal policy by including only the contribution from single-unit accident scenarios to the total mean value for selected risk metrics; and (2) a second option (Option 2) 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. Six risk metrics were selected to achieve the overall aim of this dissertation research and collectively represent three broad public health risk perspectives: (1) individual radiological health risk; (2) societal radiological health risk; and (3) societal non-radiological health risk. One figure of merit (FOM) was selected to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. This FOM is the relative contribution of multi-unit accident scenarios to the total mean value for each selected risk metric calculated under Option 2, which assumes the risk contributions from both single-unit and multi-unit accident scenarios are included.

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 State-Of-the-Art Reactor Consequence Analyses (SOARCA) pilot study\textsuperscript{16-18} was selected as the foundation for this dissertation research to build upon. This extensively peer-reviewed probabilistic consequence analysis (PCA) study developed state-of-the-art models and integrated analyses of accident progression and offsite radiological consequences to estimate 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 reactor and containment designs. These two representative NPP sites that each include two co-located reactor units were also selected as the study population and two-unit case studies for this dissertation research.

Using the state-of-the-art consequence models for each of the single-unit accident scenarios 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 was run to estimate the conditional consequence contribution to selected 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 (delay time) between concurrent two-unit accident scenarios from 1 to 7 days in one-day increments; this resulted in a total of 200 two-unit accident simulations and sets of conditional consequences for each selected risk metric.
To evaluate the effect of including the contribution to selected risk metrics from two-unit accident scenarios relative to including only the contribution from single-unit accident scenarios, efficient risk estimation models were needed to calibrate the frequency and conditional consequence results based on SOARCA models using results from previous full-scope NPP Level 3 PRA studies. This calibration is designed to estimate approximately equivalent risk results that can then be used to develop meaningful safety goal policy insights, without having to develop a resource-intensive, contemporary full-scope NPP Level 3 PRA. 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 selected risk metrics. This assumption allows for calibrating scenario-specific results to develop an adjusted measure of risk. This adjusted risk metric accounts for contributions from other potentially risk-significant categories of accident scenarios that were deliberately excluded by design from the SOARCA pilot study, and that therefore were not modeled as part of this dissertation research.

In this approach, a global average conditional probability of an accident scenario occurring in the co-located unit, given the assumed occurrence of an accident scenario in the reference unit, is used to estimate: (1) the frequency for each two-unit accident scenario; and (2) a frequency adjustment factor that is used to adjust risk measures to account for the excluded 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, which implies a 10% chance of an accident
scenario occurring in the co-located unit, given that an accident scenario occurs in
the reference unit. One-way sensitivity analyses were then performed to evaluate
the effect of varying this parameter over its entire range of possible values from 0
(no inter-unit dependence) to 1 (complete inter-unit dependence).

Although models and equations were developed specifically for a two-unit case study
involving two representative NPP sites, the overall approach and models are
generalizable and can be expanded to perform similar evaluations for NPP sites
comprised of: (1) more than two co-located units; and/or (2) other major radiological
sources (e.g., spent fuel pool units and dry cask storage facilities).
5.C. Key Conclusions

Application of the approach and models to the two-unit case studies involving two representative NPP sites generated many key findings and supplementary insights that are documented in Chapter 4. Key conclusions derived from these findings and insights are:

1. **Including the risk contribution from multi-unit accident scenarios**
   results in a non-negligible increase in the total mean value for all selected risk metrics, with multi-unit accident scenarios becoming the dominant risk contributors for higher levels of assumed inter-unit dependence. For the base case analysis, the relative contribution of two-unit accident scenarios to the total mean value was more than 10% for all selected risk metrics for each representative NPP site. This suggests that two-unit accident scenarios are non-negligible contributors to total accident risk for the representative NPP sites. For sensitivity analyses performed to evaluate the effect of variation in the assumed level of inter-unit dependence, the relative contribution of two-unit accident scenarios was more than 50% for all selected risk metrics for each representative NPP site, indicating two-unit accident scenarios are the dominant risk contributors, for assumed levels of inter-unit dependence between 50% and 100% (complete dependence).

2. **Including the risk contribution from multi-unit accident scenarios to existing safety goal QHO risk metrics would likely not impact safety-related decisions for safety goal policy applications, even under worst-case assumptions about the level of inter-unit dependence.** As expected, including the risk contribution from multi-unit accident scenarios to the safety
goal QHO risk metrics (average individual early fatality risk within one mile and average individual latent cancer fatality risk within 10 miles) reduces the available margin to each QHO. However, sufficient margin to each QHO remains to provide assurance that the residual risk level would remain acceptably low, even if multi-unit accident scenarios were considered and a worst-case assumption of complete inter-unit dependence were applied. Thus, while multi-unit accident scenarios may be significant contributors to risk metrics that measure individual risk of experiencing radiological health effects, including their contribution to the total mean value for these risk metrics is unlikely to impact safety-related decisions for applications that rely solely on the existing set of QHOs.

3. **Relying solely on risk insights for single-unit accident scenarios can lead to flawed risk management strategies for risk metrics that involve threshold effects.** Some adverse consequences arising from potential nuclear accident scenarios involve threshold effects, which means a threshold level of interest must be exceeded before any consequences are observed. Deterministic radiological health effects are a prime example, since dose thresholds for exposed tissues must be exceeded for the health effect to occur. For risk metrics that involve these types of effects, it is possible that the radiological release for a category of single-unit accident scenarios will not be substantial enough to cause threshold levels to be exceeded, leading to the conclusion that this category of accident scenarios does not contribute to risk. However, this dissertation research shows that it is possible for such a category of single-unit accident scenarios to cause threshold levels to be exceeded when combined with
concurrent accident scenarios involving one or more co-located units. Relying solely on risk insights for single-unit accident scenarios can therefore lead to flawed risk management strategies for risk metrics that involve threshold effects.

4. **Assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.** For some applications that consider multi-unit accident scenarios, a common practice is to assume that potential concurrent accident scenarios involving co-located units occur simultaneously and then—rather than explicitly modeling the radiological releases from multiple units—multiply the quantities of radionuclides released from a single unit by the number of units being considered. The underlying assumption that supports this common practice is that simultaneous releases from multiple units represents a worst-case or bounding scenario. However, for sensitivity analyses performed to evaluate the effect of variation in the assumed timing offset between concurrent accident scenarios, the FOM result for average individual latent cancer fatality risk within ten miles of the representative BWR site increased slightly as the timing offset was increased from 0 to 2 days. This counter-intuitive finding suggests that assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.

5. **Considering a broader set of public health risk metrics provides a more complete characterization of the risks to the public posed by potential nuclear accident scenarios.** Considering a broad set of public health risk metrics that go beyond measures of individual risk of experiencing radiological
health effects to include measures of societal risk for radiological and non-radiological health consequences provides a more complete characterization of the risks to the public that are attributable to potential nuclear accident scenarios involving the representative NPP sites. Although this more complete characterization cannot be used to determine whether these risks are acceptable—since quantitative objectives do not yet exist for comparison against the societal risk measures—it can provide a valuable input to current U.S. Nuclear Regulatory Commission (USNRC) and nuclear industry stakeholder deliberations about whether and to what extent the safety goal policy should be expanded to include QHOs that address societal risk.\textsuperscript{22,28}
5.D. Research Limitations and Recommendations

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 1.E. Key assumptions that underlie this study design were then highlighted in Section 1.F. Finally, Section 1.G. describes the scope of the analysis, clearly identifying aspects that have been deliberately excluded from this dissertation research.

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.

5.D.1. Set of Modeled Accident Scenarios and Efficient Risk Estimation Models

A collective total of 32 single-unit and two-unit accident scenarios (12 for the representative BWR site and 20 for the representative PWR site) were selected for detailed modeling and evaluation as part of this dissertation research. 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 including the contribution to selected risk metrics
from two-unit accident scenarios relative to including only the contribution from single-unit accident scenarios, efficient risk estimation models were developed to calibrate the frequency and conditional consequence results based on SOARCA models using results from previous full-scope NPP Level 3 PRA studies. 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 selected risk metrics. This assumption allows for calibrating scenario-specific results to develop an adjusted measure of risk that accounts for contributions from other potentially risk-significant categories of accident scenarios that were deliberately excluded by design from the SOARCA pilot study, and that therefore were not modeled as part of this dissertation research.

**Recommendation 1: Perform additional research to:** (1) **benchmark the efficient risk estimation model that uses a frequency adjustment factor to calibrate results to account for the contribution to risk from excluded accident scenarios;** 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\textsuperscript{19,21} 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. 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.

5.D.2. 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 dissertation research study. Yet these two NPP sites utilize reactor and containment designs like 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 several previous NPP Level 3 PRA studies and PCA studies.

However, this also means that these two representative 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.

Recommendation 2: Perform additional research to apply the approach and models developed as part of this dissertation 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\textsuperscript{65} described in Section 2.F.7 could serve as the foundation for a logical next step.


The existing set of QHOs used to measure attainment of the safety goals is limited to measures of individual risk for experiencing radiological health effects. Findings from this dissertation research that rely solely on results that pertain to these safety goal QHO risk metrics collectively suggest that—although multi-unit accident scenarios may be a significant source of accident risk for each two-unit representative NPP site—the level of residual risk to the public is already at an acceptably low level. Moreover, sufficient margin to each QHO exists to provide assurance that the residual risk level would remain acceptable, even if multi-unit accident scenarios were considered and a worst-case assumption of complete inter-dependence were applied. Based on these findings, it is reasonable to conclude that there would be no practical benefit to expanding the scope of the safety goal policy to include multi-unit accident scenarios, which would necessarily increase the resources needed to perform supporting NPP PRAs.

However, these findings based solely on the safety goal QHO risk metrics do not consider the effect of expanding the scope of the safety goal policy to include a broader set of public health risk metrics that address measures of societal risk for radiological and non-radiological health consequences. For this reason, additional risk-based safety goal policy insights were developed by performing a more complete characterization of risk that goes beyond the FOM results for each of the three
public health perspectives considered. Thus, this dissertation research created a body of knowledge that includes various measures of societal risk.

Although this information can inform current USNRC and nuclear industry stakeholder deliberations about whether and to what extent the safety goal policy should be expanded to include QHOs that address societal risk, it alone cannot be used to determine whether the societal risks attributable to nuclear accident scenarios involving each representative NPP site are acceptable, since quantitative objectives do not yet exist for comparison against these measures of societal risk.

**Recommendation 3.1: Gauge stakeholder interest in developing quantitative objectives for societal risk metrics.**

**Recommendation 3.2: If sufficient stakeholder interest exists, engage stakeholders to develop a set of quantitative objectives that in effect answer the question of “How safe is safe enough?” with respect to societal risk.**

**Recommendation 3.3: Perform follow-on studies that estimate societal risk metrics for comparison against developed quantitative objectives to measure attainment of the safety goals for a range of representative NPP sites.**

Examples of stakeholder groups that should participate in such efforts include: (1) public and environmental interest groups; (2) academic researchers and practitioners from multiple scientific disciplines, 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.
5.E. Additional Public Health Perspectives on Advancing Probabilistic Risk Analysis for Commercial Nuclear Power Plants

Section 5.D. identified the assumptions and limitations of this dissertation research that were judged to have the greatest potential for introducing bias or limiting the generalizability of study findings, along with recommendations for future research to address known issues. However, it is important to recognize that this dissertation research—by focusing on safety goal policy applications—addressed only a small subset of potential uses of NPP PRA studies in the regulatory and policymaking process. This section thus takes a broader view and offers additional public health perspectives on improving the quality of PRA for commercial NPPs to enhance its utility as a decision support and risk communication tool for a diverse set of stakeholder groups.

5.E.1. The Donabedian Structure-Process-Outcome Model

A widely-used framework for assessing the quality of a product, service, or activity and for designing and evaluating quality improvement initiatives—especially in the delivery of healthcare services—is Donabedian’s Structure-Process-Outcome Model.\textsuperscript{77} In this model, quality is viewed as having three sequential dimensions, with each dimension influencing the subsequent dimension(s):

1. **Structure.** The structure dimension represents the context or setting in which a product is developed, a service is provided, or an activity is performed. Structure consists of a broad range of systemic factors that influence the subsequent dimensions of quality, including: (1) legal requirements; (2) policies; (3) capabilities and resources; (4) constraints; and (5) environmental factors.
2. **Process.** The process dimension represents how a product is developed, a service is provided, or an activity is performed. Factors that comprise the structure dimension can be viewed as inputs or conditions that influence the process dimension. Process includes elements that ultimately influence outcomes, including: (1) professional codes and standards; (2) values and objectives; and (3) methods, procedures, and practices.

3. **Outcome.** The final dimension of quality is the outcome dimension. Outcomes represent the consequences that result from use of a product that has been developed, a service that has been provided, or an activity that has been performed. Outcomes of interest often relate to values and objectives, and can include consequences that are both positive and negative, as well as intended and unintended. The outcome dimension is typically the easiest and most concrete dimension to measure, and is influenced by both the structure and process dimensions. Efforts to improve outcomes can thus be directed at improving the structures and processes that influence those outcomes.


This framework for assessing and improving the quality of a product, service, or activity can be used to evaluate and identify potential opportunities for improving the use of PRA as a decision support and risk communication tool. A recent study performed by a National Research Council committee that evaluated U.S. Environmental Protection Agency (USEPA) use of risk analysis applied similar reasoning. The *Committee on Improving Risk Analysis Approaches Used by the USEPA*—which was charged with developing recommendations that could assist the
agency in producing risk analyses that are both consistent with current and evolving scientific understanding, and relevant to its many risk management missions—viewed risk analysis as both a process and the product of a process. Moreover, the committee noted that, while this dual interpretation can cause confusion, it highlights the need for attention to both the quality of the process and the quality of the product in identifying ways to improve the technical quality and utility of risk analysis as a decision support tool. In this way, the committee viewed risk analysis as a design challenge, with the goal being to develop well-designed risk analysis processes that create products that serve the needs of multiple potential end users, including: (1) risk managers and decision makers; (2) risk analysts; and (3) community and industrial stakeholders, including members of the general public.78

In its final report,78 the committee identified elements of decision situations involving public health and environmental risks that underscore the importance of process quality to developing risk analyses that are well-designed to serve the needs of various end users:

“Many decision-making situations involving matters of public health and environmental risk have five common elements: the desire to use the best scientific methods and evidence in informing decisions, uncertainty that limits the ability to characterize both the magnitude of the problem and the corresponding benefits of proposed interventions, a need for timeliness in decision-making that precludes resolving important uncertainties before decisions are required, the presence of some sort of tradeoff among disparate adverse outcomes (which may be health, ecologic, or economic outcomes, each affecting a different set of stakeholders), and the reality that, because of the inherent complexity of the systems being managed and the long-term implications of many decisions (such as cancer latency, changes in the structure of ecosystems, or multiple simultaneous sources of exposure), there will be little or no short-term feedback as to whether the desired outcome has been achieved by the decisions.
The combination of uncertainty in the scientific data and assumptions (the “inputs”) and inability to validate assessment results directly or to isolate and evaluate the impact of a resulting decision (the “outputs”) creates a situation in which decision-makers, the scientific community, the public, industry and other stakeholders have little choice but to rely on the overall quality of the many processes used in the conduct of risk assessment to provide some assurance that the assessment is aligned with societal goals...

...Given the demands of health and environmental decision-making, perhaps the most appropriate element of quality in risk-assessment products is captured in their ability to improve the capacity of decision-makers to make informed decisions in the presence of substantial, inevitable and irreducible uncertainty. A secondary but surely important quality is the ability of the assessment products to improve other stakeholders’ understanding and to foster and support the broader public interests in the quality of the decision-making process (for example, fairness, transparency, and efficiency). Those attributes are difficult to measure, and some elements of quality often cannot be judged until some time after the completion of the risk assessment.” (pp. 65-67)

These attributes of public health and environmental risk decisions also apply to risk management or safety-related decisions involving commercial NPPs. This suggests that efforts to identify potential areas for advancing PRA for commercial NPPs should focus on the quality of the USNRC’s PRA processes. Moreover, according to Donabedian’s Structure-Process-Outcome Model,77 such efforts should also focus on the quality of the structural elements that influence PRA processes.


Relying on some insights from the Committee on Improving Risk Analysis Approaches Used by the USEPA,78 Donabedian’s Structure-Process-Outcome Model77 is used to identify questions for future research that could improve the technical quality and utility of NPP PRA as a decision support and risk communication tool for a diverse set of stakeholder groups.
5.E.3.I. Questions Related to Structure

- How does the USNRC’s interpretation of its statutory responsibilities under the Atomic Energy Act of 1954, as amended (AEA), influence the scope of the safety goal policy and its PRA processes? The USNRC has interpreted its legislative mandate to protect public health and safety to mean that it is required to provide reasonable assurance that the public is adequately protected from the direct radiological health effects that can be caused by exposure to ionizing radiation from routine or accidental releases of radioactive materials from USNRC-licensed nuclear facilities. Although the safety goal policy pertains to the agency’s broad discretionary authority under the AEA to impose additional requirements that go beyond ensuring the public is adequately protected, the existing QHOs that are limited to measures of individual risk of dying from acute or chronic accident radiation exposure suggest that the agency’s relatively narrow interpretation of its legislative mandate may have constrained the scope of the safety goal policy and its PRA processes. From a public health systems perspective, there are multiple determinants that impact public health at different levels of the system, including: (1) innate individual traits (e.g., age, sex, race, biological factors); (2) individual behavior (e.g., lifestyle factors); (3) social, family, and community networks; (4) living and working conditions (e.g., psychosocial factors, socioeconomic status, natural and built environment, public health and medical services); and (5) broad social, economic, cultural, health, and environmental conditions and policies at the global, national, state, and local levels. Experience with real-world nuclear accidents has shown that the dominant public health consequences attributable to such accidents arise from
their impact on many of these determinants. What effect would expanding the scope of the safety goal policy and PRA processes to include consideration of these impacts have on regulatory decision-making at the USNRC and its stakeholders?

- **What effect would using alternative summary statistics have on regulatory decision-making?** As a matter of policy, the USNRC uses the distribution mean for probabilistic risk metrics when comparing PRA results to established criteria for decision-making purposes. Other agencies with public health protection functions (e.g., USEPA) often rely on the use of summary statistics that are generally more conservative than the mean (e.g., the 95th percentile) when comparing risk results to established decision criteria. Would use of a more conservative summary statistic that—if biased, would err on the side of being more protective of the public’s health—result in different regulatory decisions?

5.E.3.II. **Questions Related to Process**

- **Are there any improvements that can be made to stakeholder participation in the PRA process at the USNRC?** Decision-making processes should strive to be inclusive with respect to the participation of affected and interested parties. PRA processes should thus be structured to accommodate the needs of diverse stakeholders, including: (1) accepting stakeholder input at appropriate points as part of an iterative analytic-deliberative process; (2) ensuring fairness in their influence on various aspects of the design of the PRA process and products; (3) facilitating their desired level of understanding of the process; and (4) meeting their specific information needs.
Although the USNRC conducts public meetings that provide stakeholders with opportunities to engage in various decision-making and PRA processes, there are likely ways to improve upon these participatory efforts that can enhance both the technical quality and utility of PRA as a decision support and risk communication tool.

- **How should low-probability/high-consequence accident scenarios that dominate risk in the tails of probability distributions be treated in NPP PRAs and how should the risk contributions from these accident scenarios be presented and displayed to facilitate decision maker and stakeholder understanding?** Performing a NPP PRA requires making choices about criteria that will be used to screen out or exclude classes of accident scenarios from the PRA model. In practice, this is typically done by establishing a frequency cutoff, such that all accident scenarios with an estimated frequency below that level are excluded from the PRA model, regardless of their potential conditional consequences. What is the right level for that frequency cutoff? How unlikely is so unlikely as to be practically impossible from a PRA perspective? In addition, for those low-probability/high-consequence accident scenarios that are included in the model, how should the risk contributions be presented and displayed to facilitate decision maker and stakeholder understanding? These low-probability/high-consequence accident scenarios typically dominate risk in the tails of the probability distributions and can significantly influence the mean value.
5.F. Chapter Summary and Conclusion

This chapter provided a high-level summary of the research that was performed and then presented the key conclusions derived from the findings and insights developed from the two-unit case studies involving two representative NPP sites. These key conclusions were:

1. Including the risk contribution from multi-unit accident scenarios results in a non-negligible increase in the total mean value for all selected risk metrics, with multi-unit accident scenarios becoming the dominant risk contributors for higher levels of assumed inter-unit dependence.

2. Even under worst-case assumptions about the level of inter-unit dependence, including the risk contribution from multi-unit accident scenarios to existing safety goal QHO risk metrics that are limited to measures of average individual risk of radiological health effects would likely not impact safety-related decisions for safety goal policy applications.

3. Relying solely on risk insights for single-unit accident scenarios can lead to flawed risk management strategies for risk metrics that involve threshold effects.

4. Assuming potential concurrent accident scenarios occur simultaneously may not be a conservative or bounding assumption for all risk metrics of interest.

5. Considering a broader set of public health risk metrics provides a more complete characterization of the risks to the public posed by potential nuclear accident scenarios.
Next, the chapter identified the assumptions and limitations in study design that were judged to have the greatest potential for introducing bias or limiting the generalizability of study findings, along with recommendations for future research to address known issues. These recommendations for future research were to:

1. Benchmark the efficient risk estimation model that uses a frequency adjustment factor to calibrate results to account for the contribution to risk from excluded accident scenarios.
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.
3. Apply the approach and models developed as part of this dissertation research to additional reactor and containment designs and/or NPP sites.
4. Gauge stakeholder interest in developing quantitative objectives for societal risk metrics and, if sufficient stakeholder interest exists, engage stakeholders to develop a set of quantitative objectives that in effect answer the question of “How safe is safe enough?” with respect to societal risk.
5. Perform follow-on studies that estimate societal risk metrics for comparison against developed quantitative objectives to measure safety goal attainment for a range of representative NPP sites.

Finally, the chapter concluded with some additional public health perspectives on improving the quality of PRA for commercial NPPs to enhance its utility as a decision support and risk communication tool for a diverse set of stakeholder groups.
Although there are limitations to consider, this dissertation research makes three significant contributions to the literature:

1. It specifies and applies efficient models for estimating the contributions to selected risk metrics from categories of single-unit and multi-unit accident scenarios using state-of-the-art consequence models from a contemporary PCA study that leveraged decades of severe accident research and advanced analytical tools to develop realistic estimates of the offsite radiological consequences attributable to important nuclear accident scenarios.

2. It develops and evaluates state-of-the-art consequence models for concurrent multi-unit accident scenarios involving both: (1) simultaneous accident scenarios involving multiple co-located reactor units; and (2) staggered accident scenarios in which the timing offset (delay time) 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 selected risk metrics; and (2) the effects of expanding the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics that portray a more complete picture of the public health risks attributable to potential single-unit and multi-unit accident scenarios.

Insights derived from this dissertation 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: (1) consideration of the risk contribution from multi-unit accident scenarios; and (2)
QHOs that address societal risk.\textsuperscript{22,28} Regardless of the outcome of these deliberations, this dissertation research contributes to a growing body of evidence that challenges the state of knowledge about risks to the public posed by potential nuclear accident scenarios that existed when the safety goal policy was originally developed. While radiological health effects were understandably the primary public health concern after the emergence of nuclear power technology, our understanding of nuclear accident consequences has evolved considerably through: (1) accumulation of operational experience over several decades of NPP operations, including lessons learned from real-world nuclear accidents; (2) decades of research involving severe nuclear accident phenomena and NPP systems performance under severe accident conditions; and (3) development of advanced modeling and analytical tools that have been applied in contemporary NPP PRA and PCA studies. We now know that societal disruption caused in part by implementing protective actions to avert radiological dose, and the indirect effects on public health that stem from this disruption, represent the most important adverse consequences from nuclear accidents. Yet the existing safety goal policy and QHOs used to evaluate proposed regulatory actions that aim to further enhance NPP safety beyond the minimum level needed to ensure adequate protection do not account for these types of adverse public health consequences. Under the existing policy, proposed regulatory actions that aim to reduce the risks to the public from accidents like the one that occurred at Fukushima can be rejected because individual radiological health risks are deemed to be acceptably low. As a society, it may be time to reevaluate what we know about the public risks from potential nuclear accident scenarios and determine whether this system of managing public health and safety risks remains acceptable.
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Definition</th>
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<tbody>
<tr>
<td>AC</td>
<td>alternating current</td>
</tr>
<tr>
<td>ACRS</td>
<td>Advisory Committee on Reactor Safeguards</td>
</tr>
<tr>
<td>AEA</td>
<td>Atomic Energy Act of 1954, as amended</td>
</tr>
<tr>
<td>AEC</td>
<td>U.S. Atomic Energy Commission</td>
</tr>
<tr>
<td>ARS</td>
<td>acute radiation syndrome</td>
</tr>
<tr>
<td>ATD</td>
<td>atmospheric transport and dispersion</td>
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<tr>
<td>BDBA</td>
<td>beyond-design-basis accident</td>
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<tr>
<td>BWR</td>
<td>boiling-water reactor</td>
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<tr>
<td>CCF</td>
<td>common-cause failure</td>
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<tr>
<td>CDC</td>
<td>U.S. Centers for Disease Control and Prevention</td>
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<td>CDF</td>
<td>core damage frequency</td>
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<td>Cs-137</td>
<td>Cesium-137</td>
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<td>design-basis threat</td>
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<td>DC</td>
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<td>EDG</td>
<td>emergency diesel generator</td>
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<td>EDMG</td>
<td>extensive damage mitigation guideline</td>
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<td>EOP</td>
<td>emergency operating procedure</td>
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<td>EPZ</td>
<td>emergency planning zone</td>
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<td>ERDA</td>
<td>Energy Research and Development Administration</td>
</tr>
<tr>
<td>ERP</td>
<td>emergency response plan</td>
</tr>
<tr>
<td>ETE</td>
<td>evacuation time estimate</td>
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FDA    U.S. Food and Drug Administration
FDNPS  Fukushima Daiichi Nuclear Power Station
FOM    figure of merit
GI     generic issue
GIP    Generic Issues Program
HRA    human reliability analysis
INES   International Nuclear and Radiological Event Scale
ISLOCA interfacing systems loss-of-coolant accident
I-131  Iodine-131
KI     potassium iodide
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
MELCOR severe nuclear accident progression analysis code (not an acronym)
MelMACCS MELCOR-to-MACCS interface software utility
MGL    multiple Greek letter
NPP    nuclear power plant
NPV    net present value
PAG    protective action guide
PCA    probabilistic consequence analysis
<table>
<thead>
<tr>
<th>Acronym</th>
<th>Full Form</th>
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<tbody>
<tr>
<td>PRA</td>
<td>probabilistic risk analysis</td>
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<td>PTSD</td>
<td>post-traumatic stress disorder</td>
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<tr>
<td>PWR</td>
<td>pressurized-water reactor</td>
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<tr>
<td>QHO</td>
<td>quantitative health objective</td>
</tr>
<tr>
<td>ORO</td>
<td>offsite response organization</td>
</tr>
<tr>
<td>RCIC</td>
<td>Reactor Core Isolation Cooling System</td>
</tr>
<tr>
<td>SAMA</td>
<td>severe accident mitigation alternative</td>
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<td>SAMDA</td>
<td>severe accident mitigation design alternative</td>
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<td>SAMG</td>
<td>severe accident mitigation guideline</td>
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<td>SOARCA</td>
<td>State-Of-the-Art Reactor Consequence Analyses</td>
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<td>SSC</td>
<td>structure, system, component</td>
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<td>STSBO</td>
<td>short-term station blackout</td>
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<tr>
<td>TISGTR</td>
<td>thermally-induced steam generator tube rupture</td>
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<td>USEPA</td>
<td>U.S. Environmental Protection Agency</td>
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<tr>
<td>USNRC</td>
<td>U.S. Nuclear Regulatory Commission</td>
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Appendix II. Results Tables for the Representative Boiling-Water Reactor Site (Peach Bottom)

II.A. Notes for Results Tables

1. As shown in Equation (1) and Equation (6) in Chapter 3, mean unadjusted risk for each accident scenario is the product of its mean frequency and its mean conditional consequence. As shown in Equation (3) and Equation (8) in Chapter 3, mean adjusted risk for each accident scenario is then the product of its mean unadjusted risk and a frequency adjustment factor used to calibrate the mean unadjusted risk to results from a previous full-scope Level 3 probabilistic risk analysis (PRA) for the nuclear power plant (NPP) site to account for the frequency contribution from potential accident scenarios that have not been modeled as part of this study. Based on site-specific results from the NUREG-1150 study, the frequency adjustment factor for the representative boiling-water reactor (BWR) site was estimated to be 7.67 using Equation (2) and Equation (7) in Chapter 3.

2. The set of modeled single-unit and two-unit accident scenarios are defined to be mutually exclusive, meaning they cannot jointly occur. For this reason, a summation of the mean conditional consequences across all single-unit accident scenarios and across all two-unit accident scenarios would be inappropriate and meaningless. The total (sum) value for the mean conditional consequences is therefore designated as not applicable (N/A) for both single-unit and two-unit accident scenarios for all selected risk metrics.
3. Under this mutually exclusive condition, the total contribution to each selected risk metric from all single-unit accident scenarios is estimated by summing the scenario-specific mean adjusted risk values across all single-unit accident scenarios using Equation (4) in Chapter 3. Similarly, the total contribution to each selected risk metric from all two-unit accident scenarios is estimated by summing the scenario-specific mean adjusted risk values across all two-unit accident scenarios using Equation (9) in Chapter 3.

4. The relative contribution of each accident scenario to total mean risk is calculated by dividing its mean adjusted risk by the total mean risk under Option 2, which includes the total contribution to each selected risk metric from both single-unit and two-unit accident scenarios. This total mean risk under Option 2 is calculated by summing the total mean adjusted risk values for single-unit and two-unit accident scenarios using Equation (11) in Chapter 3.

5. The total relative contribution of single-unit accident scenarios to total mean risk is calculated by dividing the total contribution to each selected risk metric from all single-unit accident scenarios by the total mean risk under Option 2. Similarly, the total relative contribution of two-unit accident scenarios to total mean risk is calculated by dividing the total contribution to each selected risk metric from all two-unit accident scenarios by the total mean risk under Option 2. For two-unit accident scenarios, this total relative contribution represents the figure of merit (FOM) used to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. This FOM is calculated using Equation (16) in Chapter 3.
II.B. Results for Individual Radiological Health Risk Perspective
Table II-I. Representative BWR Site Results for Average Individual Early Fatality Risk (0-1 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (dimensionless)</th>
<th>Mean Adjusted Risk (per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>2.8E-07</td>
<td>6.5E-13</td>
<td>39%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>2.8E-05</td>
<td>N/A [Note 2]</td>
<td>6.5E-13 [Note 3]</td>
<td>39% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>6.8E-07</td>
<td>2.6E-13</td>
<td>16%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>6.8E-07</td>
<td>2.6E-13</td>
<td>16%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>1.1E-05</td>
<td>4.2E-13</td>
<td>25%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>7.4E-07</td>
<td>2.8E-14</td>
<td>2%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>7.4E-07</td>
<td>2.8E-14</td>
<td>2%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>5.5E-06</td>
<td>N/A [Note 2]</td>
<td>1.0E-12 [Note 3]</td>
<td>61% [Note 5]</td>
</tr>
</tbody>
</table>
Table II-II. Representative BWR Site Results for Average Individual Latent Cancer Fatality Risk (0-10 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (dimensionless)</th>
<th>Mean Adjusted Risk (per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>9.2E-05</td>
<td>2.1E-09</td>
<td>59%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>2.1E-04</td>
<td>4.8E-10</td>
<td>13%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>7.3E-05</td>
<td>1.7E-10</td>
<td>5%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>2.8E-05</td>
<td>N/A [Note 2]</td>
<td>2.8E-09 [Note 3]</td>
<td>77% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>1.3E-04</td>
<td>5.1E-10</td>
<td>14%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>2.3E-04</td>
<td>8.9E-11</td>
<td>2%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>1.2E-04</td>
<td>4.8E-11</td>
<td>1%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>2.3E-04</td>
<td>8.9E-11</td>
<td>2%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>2.8E-04</td>
<td>1.1E-11</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>2.3E-04</td>
<td>8.8E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>1.2E-04</td>
<td>4.8E-11</td>
<td>1%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>2.3E-04</td>
<td>8.8E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>1.0E-04</td>
<td>4.0E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>5.5E-06</td>
<td>N/A [Note 2]</td>
<td>8.1E-10 [Note 3]</td>
<td>23% [Note 5]</td>
</tr>
</tbody>
</table>
II.C. Results for Societal Radiological Health Risk Perspective
<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (early fatalities)</th>
<th>Mean Adjusted Risk (early fatalities per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>1.4E-05</td>
<td>3.1E-11</td>
<td>36%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>2.8E-05</td>
<td>N/A [Note 2]</td>
<td>3.1E-11 [Note 3]</td>
<td>36% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>3.7E-05</td>
<td>1.4E-11</td>
<td>17%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>3.7E-05</td>
<td>1.4E-11</td>
<td>17%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>5.9E-04</td>
<td>2.3E-11</td>
<td>27%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>4.0E-05</td>
<td>1.5E-12</td>
<td>2%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>4.0E-05</td>
<td>1.5E-12</td>
<td>2%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>5.5E-06</td>
<td>N/A [Note 2]</td>
<td>5.4E-11 [Note 3]</td>
<td>64% [Note 5]</td>
</tr>
</tbody>
</table>
Table II-IV. Representative BWR Site Results for Total Latent Cancer Fatality Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (latent cancer fatalities)</th>
<th>Mean Adjusted Risk (latent cancer fatalities per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>1.5E+02</td>
<td>3.5E-03</td>
<td>41%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>1.1E+03</td>
<td>2.5E-03</td>
<td>29%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>1.3E+02</td>
<td>3.1E-04</td>
<td>4%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>2.8E-05</td>
<td>N/A [Note 2]</td>
<td>6.3E-03 [Note 3]</td>
<td>74% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>2.6E+02</td>
<td>9.8E-04</td>
<td>11%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>1.2E+03</td>
<td>4.5E-04</td>
<td>5%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>2.5E+02</td>
<td>9.6E-05</td>
<td>1%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>1.2E+03</td>
<td>4.5E-04</td>
<td>5%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>2.2E+03</td>
<td>8.3E-05</td>
<td>1%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>1.2E+03</td>
<td>4.5E-05</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>2.5E+02</td>
<td>9.6E-05</td>
<td>1%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>1.2E+03</td>
<td>4.5E-05</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>2.2E+02</td>
<td>8.6E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>5.5E-06</td>
<td>N/A [Note 2]</td>
<td>2.3E-03 [Note 3]</td>
<td>26% [Note 5]</td>
</tr>
</tbody>
</table>
II.D. Results for Societal Non-Radiological Health Risk Perspective
Table II-V. Representative BWR Site Results for Total Emergency Phase Population Relocation Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (relocated individuals)</th>
<th>Mean Adjusted Risk (relocated individuals per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>2.0E+05</td>
<td>4.6E+00</td>
<td>51%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>7.9E+05</td>
<td>1.8E+00</td>
<td>20%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>1.9E+05</td>
<td>4.4E-01</td>
<td>5%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td><strong>2.8E-05</strong></td>
<td>N/A [Note 2]</td>
<td><strong>6.9E+00</strong> [Note 3]</td>
<td><strong>76%</strong> [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>3.1E+05</td>
<td>1.2E+00</td>
<td>13%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>7.9E+05</td>
<td>3.0E-01</td>
<td>3%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>2.9E+05</td>
<td>1.1E-01</td>
<td>1%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>7.9E+05</td>
<td>3.0E-01</td>
<td>3%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>9.9E+05</td>
<td>3.8E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>7.9E+05</td>
<td>3.0E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>2.9E+05</td>
<td>1.1E-01</td>
<td>1%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>7.9E+05</td>
<td>3.0E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>2.8E+05</td>
<td>1.1E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td><strong>5.5E-06</strong></td>
<td>N/A [Note 2]</td>
<td><strong>2.1E+00</strong> [Note 3]</td>
<td><strong>24%</strong> [Note 5]</td>
</tr>
</tbody>
</table>
Table II-VI. Representative BWR Site Results for Total Recovery Phase Population Relocation Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (relocated individuals)</th>
<th>Mean Adjusted Risk (relocated individuals per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>2.3E-05</td>
<td>3.0E+04</td>
<td>6.9E-01</td>
<td>30%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>2.3E-06</td>
<td>4.2E+05</td>
<td>9.6E-01</td>
<td>42%</td>
</tr>
<tr>
<td>STSBO-RCIC</td>
<td>2.3E-06</td>
<td>3.4E+04</td>
<td>7.9E-02</td>
<td>3%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>2.8E-05</td>
<td>N/A [Note 2]</td>
<td>1.7E+00 [Note 3]</td>
<td>75% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>BWR1 - Unit 1: LTSBO, Unit 2: LTSBO</td>
<td>3.8E-06</td>
<td>4.4E+04</td>
<td>1.7E-01</td>
<td>7%</td>
</tr>
<tr>
<td>BWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>3.8E-07</td>
<td>4.2E+05</td>
<td>1.6E-01</td>
<td>7%</td>
</tr>
<tr>
<td>BWR3 - Unit 1: LTSBO, Unit 2: STSBO-RCIC</td>
<td>3.8E-07</td>
<td>4.2E+04</td>
<td>1.7E-02</td>
<td>1%</td>
</tr>
<tr>
<td>BWR4 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>4.2E+05</td>
<td>1.6E-01</td>
<td>7%</td>
</tr>
<tr>
<td>BWR5 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>4.2E+05</td>
<td>1.6E-02</td>
<td>1%</td>
</tr>
<tr>
<td>BWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>4.2E+05</td>
<td>1.6E-02</td>
<td>1%</td>
</tr>
<tr>
<td>BWR7 - Unit 1: STSBO-RCIC, Unit 2: LTSBO</td>
<td>3.8E-07</td>
<td>4.4E+04</td>
<td>1.7E-02</td>
<td>1%</td>
</tr>
<tr>
<td>BWR8 - Unit 1: STSBO-RCIC, Unit 2: STSBO-Base</td>
<td>3.8E-08</td>
<td>4.2E+05</td>
<td>1.6E-02</td>
<td>1%</td>
</tr>
<tr>
<td>BWR9 - Unit 1: STSBO-RCIC, Unit 2: STSBO-RCIC</td>
<td>3.8E-08</td>
<td>4.8E+04</td>
<td>1.8E-03</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>5.5E-06</td>
<td>N/A [Note 2]</td>
<td>5.7E-01 [Note 3]</td>
<td>25% [Note 5]</td>
</tr>
</tbody>
</table>
Appendix III. Results Tables for the Representative Pressurized-Water Reactor Site (Surry)

III.A. Notes for Results Tables

1. As shown in Equation (1) and Equation (6) in Chapter 3, mean unadjusted risk for each accident scenario is the product of its mean frequency and its mean conditional consequence. As shown in Equation (3) and Equation (8) in Chapter 3, mean adjusted risk for each accident scenario is then the product of its mean unadjusted risk and a frequency adjustment factor used to calibrate the mean unadjusted risk to results from a previous full-scope Level 3 probabilistic risk analysis (PRA) for the nuclear power plant (NPP) site to account for the frequency contribution from potential accident scenarios that have not been modeled as part of this study. Based on site-specific results from the NUREG-1150 study, the frequency adjustment factor for the representative pressurized-water reactor (PWR) site was estimated to be 3.39 using Equation (2) and Equation (7) in Chapter 3.

2. The set of modeled single-unit and two-unit accident scenarios are defined to be mutually exclusive, meaning they cannot jointly occur. For this reason, a summation of the mean conditional consequences across all single-unit accident scenarios and across all two-unit accident scenarios would be inappropriate and meaningless. The total (sum) value for the mean conditional consequences is therefore designated as not applicable (N/A) for both single-unit and two-unit accident scenarios for all selected risk metrics.
3. Under this mutually exclusive condition, the total contribution to each selected risk metric from all single-unit accident scenarios is estimated by summing the scenario-specific mean adjusted risk values across all single-unit accident scenarios using Equation (4) in Chapter 3. Similarly, the total contribution to each selected risk metric from all two-unit accident scenarios is estimated by summing the scenario-specific mean adjusted risk values across all two-unit accident scenarios using Equation (9) in Chapter 3.

4. The relative contribution of each accident scenario to total mean risk is calculated by dividing its mean adjusted risk by the total mean risk under Option 2, which includes the total contribution to each selected risk metric from both single-unit and two-unit accident scenarios. This total mean risk under Option 2 is calculated by summing the total mean adjusted risk values for single-unit and two-unit accident scenarios using Equation (11) in Chapter 3.

5. The total relative contribution of single-unit accident scenarios to total mean risk is calculated by dividing the total contribution to each selected risk metric from all single-unit accident scenarios by the total mean risk under Option 2. Similarly, the total relative contribution of two-unit accident scenarios to total mean risk is calculated by dividing the total contribution to each selected risk metric from all two-unit accident scenarios by the total mean risk under Option 2. For two-unit accident scenarios, this total relative contribution represents the figure of merit (FOM) used to evaluate the effects of expanding the scope of the safety goal policy to include the risk contribution from multi-unit accident scenarios and a broader set of public health risk metrics. This FOM is calculated using Equation (16) in Chapter 3.
III.B. Results for Individual Radiological Health Risk Perspective
<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (dimensionless)</th>
<th>Mean Adjusted Risk (per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>1.7E-04</td>
<td>1.8E-11</td>
<td>72%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>7.6E-05</td>
<td>N/A [Note 2]</td>
<td>1.8E-11 [Note 3]</td>
<td>72%</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>1.7E-04</td>
<td>3.1E-12</td>
<td>13%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>1.7E-04</td>
<td>3.1E-13</td>
<td>1%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>5.8E-10</td>
<td>2.8E-18</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>1.7E-04</td>
<td>6.2E-14</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>1.7E-04</td>
<td>3.1E-12</td>
<td>13%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>1.7E-04</td>
<td>3.1E-13</td>
<td>1%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>1.7E-04</td>
<td>6.2E-14</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>7.5E-04</td>
<td>2.0E-14</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>1.5E-05</td>
<td>N/A [Note 2]</td>
<td>6.9E-12 [Note 3]</td>
<td>28%</td>
</tr>
</tbody>
</table>
Table III-II. Representative PWR Site Results for Average Individual Latent Cancer Fatality Risk (0-10 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (dimensionless)</th>
<th>Mean Adjusted Risk (per year)</th>
<th>Relative Contribution to Total Mean Risk</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>4.7E-05</td>
<td>3.2E-09</td>
<td>54%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>9.5E-05</td>
<td>6.4E-10</td>
<td>11%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>3.2E-04</td>
<td>4.4E-10</td>
<td>7%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>3.0E-04</td>
<td>3.0E-11</td>
<td>1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>7.6E-05</td>
<td>N/A [Note 2]</td>
<td>4.3E-09 [Note 3]</td>
<td>73% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>8.6E-05</td>
<td>1.0E-09</td>
<td>18%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>1.3E-04</td>
<td>1.6E-10</td>
<td>3%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>3.6E-04</td>
<td>8.6E-11</td>
<td>1%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>3.3E-04</td>
<td>6.0E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>1.3E-04</td>
<td>1.6E-10</td>
<td>3%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>1.7E-04</td>
<td>2.1E-11</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>3.9E-04</td>
<td>9.5E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>3.7E-04</td>
<td>6.7E-13</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>3.6E-04</td>
<td>8.6E-11</td>
<td>1%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>3.9E-04</td>
<td>9.5E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>4.4E-04</td>
<td>2.1E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>5.6E-04</td>
<td>2.0E-13</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>3.3E-04</td>
<td>6.0E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>3.7E-04</td>
<td>6.7E-13</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>5.6E-04</td>
<td>2.0E-13</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>4.2E-04</td>
<td>1.1E-14</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>1.5E-05</td>
<td>N/A [Note 2]</td>
<td>1.6E-09 [Note 3]</td>
<td>27% [Note 5]</td>
</tr>
</tbody>
</table>
III.C. Results for Societal Radiological Health Risk Perspective
Table III-III. Representative PWR Site Results for Total Early Fatality Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (early fatalities)</th>
<th>Mean Adjusted Risk (early fatalities per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>6.9E-03</td>
<td>7.0E-10</td>
<td>71%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>7.6E-05</td>
<td>N/A [Note 2]</td>
<td>7.0E-10 [Note 3]</td>
<td>71% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>6.9E-03</td>
<td>1.3E-10</td>
<td>13%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>6.9E-03</td>
<td>1.3E-11</td>
<td>1%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>0.0E+00</td>
<td>0.0E+00</td>
<td>0%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>2.3E-08</td>
<td>1.1E-16</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>6.9E-03</td>
<td>2.5E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>6.9E-03</td>
<td>1.3E-10</td>
<td>13%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>6.9E-03</td>
<td>1.3E-11</td>
<td>1%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>6.9E-03</td>
<td>2.5E-12</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>2.5E-01</td>
<td>6.8E-12</td>
<td>1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>1.5E-05</td>
<td>N/A [Note 2]</td>
<td>2.8E-10 [Note 3]</td>
<td>29% [Note 5]</td>
</tr>
</tbody>
</table>
Table III-IV. Representative PWR Site Results for Total Latent Cancer Fatality Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (latent cancer fatalities)</th>
<th>Mean Adjusted Risk (latent cancer fatalities per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Single-Unit Accident Scenarios</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>1.8E+01</td>
<td>1.2E-03</td>
<td>53%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>3.3E+01</td>
<td>2.3E-04</td>
<td>10%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>1.5E+02</td>
<td>2.0E-04</td>
<td>9%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>3.9E+02</td>
<td>3.9E-05</td>
<td>2%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>7.6E-05</td>
<td>N/A [Note 2]</td>
<td>1.7E-03 [Note 3]</td>
<td>73% [Note 5]</td>
</tr>
<tr>
<td>Two-Unit Accident Scenarios</td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>3.4E+01</td>
<td>4.1E-04</td>
<td>18%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>4.9E+01</td>
<td>5.9E-05</td>
<td>3%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>1.6E+02</td>
<td>3.9E-05</td>
<td>2%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>4.0E+02</td>
<td>7.2E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>4.9E+01</td>
<td>5.9E-05</td>
<td>3%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>6.3E+01</td>
<td>7.6E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>1.8E+02</td>
<td>4.3E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>4.1E+02</td>
<td>7.4E-07</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>1.6E+02</td>
<td>3.9E-05</td>
<td>2%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>1.8E+02</td>
<td>4.3E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>2.5E+02</td>
<td>1.2E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>5.0E+02</td>
<td>1.8E-07</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>4.0E+02</td>
<td>7.2E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>4.1E+02</td>
<td>7.4E-07</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>5.0E+02</td>
<td>1.8E-07</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>6.4E+02</td>
<td>1.7E-08</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>Total (Sum)</td>
<td>1.5E-05</td>
<td>N/A [Note 2]</td>
<td>6.5E-04 [Note 3]</td>
<td>27% [Note 5]</td>
</tr>
</tbody>
</table>
III.D. Results for Societal Non-Radiological Health Risk Perspective
Table III-V. Representative PWR Site Results for Total Emergency Phase Population Relocation Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (relocated individuals)</th>
<th>Mean Adjusted Risk (relocated individuals per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>1.8E+05</td>
<td>1.2E+01</td>
<td>74%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>1.8E+05</td>
<td>1.2E+00</td>
<td>7%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>1.9E+05</td>
<td>2.5E-01</td>
<td>2%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>2.8E+05</td>
<td>2.9E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>7.6E-05</td>
<td>N/A [Note 2]</td>
<td>1.4E+01 [Note 3]</td>
<td>83% [Note 5]</td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>1.8E+05</td>
<td>2.2E+00</td>
<td>13%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>1.8E+05</td>
<td>2.2E-01</td>
<td>1%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>1.9E+05</td>
<td>4.5E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>2.8E+05</td>
<td>5.1E-03</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>1.8E+05</td>
<td>2.2E-01</td>
<td>1%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>1.8E+05</td>
<td>2.2E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>1.9E+05</td>
<td>4.5E-03</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>2.8E+05</td>
<td>5.1E-04</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>1.9E+05</td>
<td>4.5E-02</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>1.9E+05</td>
<td>4.5E-03</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>1.9E+05</td>
<td>9.4E-04</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>2.8E+05</td>
<td>1.0E-04</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>2.8E+05</td>
<td>5.1E-03</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>2.8E+05</td>
<td>5.1E-04</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>2.8E+05</td>
<td>1.0E-04</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>3.1E+05</td>
<td>8.4E-06</td>
<td>&lt;1%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td>1.5E-05</td>
<td>N/A [Note 2]</td>
<td>2.8E+00 [Note 3]</td>
<td>17% [Note 5]</td>
</tr>
</tbody>
</table>
### Table III-VI. Representative PWR Site Results for Total Recovery Phase Population Relocation Risk (0-50 mi)

<table>
<thead>
<tr>
<th>Accident Scenario</th>
<th>Mean Frequency (per year)</th>
<th>Mean Conditional Consequence (relocated individuals)</th>
<th>Mean Adjusted Risk (relocated individuals per year) [Note 1]</th>
<th>Relative Contribution to Total Mean Risk [Note 4]</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Single-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>LTSBO</td>
<td>6.8E-05</td>
<td>1.1E+00</td>
<td>7.5E-05</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>STSBO-Base</td>
<td>6.8E-06</td>
<td>9.8E+00</td>
<td>6.6E-05</td>
<td>&lt;1%</td>
</tr>
<tr>
<td>STSBO-TISGTR</td>
<td>1.4E-06</td>
<td>6.4E+03</td>
<td>8.7E-03</td>
<td>52%</td>
</tr>
<tr>
<td>ISLOCA</td>
<td>1.0E-07</td>
<td>3.0E+04</td>
<td>3.1E-03</td>
<td>19%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td><strong>7.6E-05</strong></td>
<td><strong>N/A [Note 2]</strong></td>
<td><strong>1.2E-02 [Note 3]</strong></td>
<td><strong>71% [Note 5]</strong></td>
</tr>
<tr>
<td><strong>Two-Unit Accident Scenarios</strong></td>
<td></td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>PWR1 - Unit 1 LTSBO, Unit 2: LTSBO</td>
<td>1.2E-05</td>
<td>6.1E+00</td>
<td>7.4E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR2 - Unit 1: LTSBO, Unit 2: STSBO-Base</td>
<td>1.2E-06</td>
<td>1.0E+01</td>
<td>1.3E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR3 - Unit 1: LTSBO, Unit 2: STSBO-TISGTR</td>
<td>2.4E-07</td>
<td>6.0E+03</td>
<td>1.5E-03</td>
<td>9%</td>
</tr>
<tr>
<td>PWR4 - Unit 1: LTSBO, Unit 2: ISLOCA</td>
<td>1.8E-08</td>
<td>2.7E+04</td>
<td>4.9E-04</td>
<td>3%</td>
</tr>
<tr>
<td>PWR5 - Unit 1: STSBO-Base, Unit 2: LTSBO</td>
<td>1.2E-06</td>
<td>1.0E+01</td>
<td>1.3E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR6 - Unit 1: STSBO-Base, Unit 2: STSBO-Base</td>
<td>1.2E-07</td>
<td>8.6E+01</td>
<td>1.0E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR7 - Unit 1: STSBO-Base, Unit 2: STSBO-TISGTR</td>
<td>2.4E-08</td>
<td>6.1E+03</td>
<td>1.5E-04</td>
<td>1%</td>
</tr>
<tr>
<td>PWR8 - Unit 1: STSBO-Base, Unit 2: ISLOCA</td>
<td>1.8E-09</td>
<td>2.7E+04</td>
<td>4.9E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR9 - Unit 1: STSBO-TISGTR, Unit 2: LTSBO</td>
<td>2.4E-07</td>
<td>6.0E+03</td>
<td>1.5E-03</td>
<td>9%</td>
</tr>
<tr>
<td>PWR10 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-Base</td>
<td>2.4E-08</td>
<td>6.1E+03</td>
<td>1.5E-04</td>
<td>1%</td>
</tr>
<tr>
<td>PWR11 - Unit 1: STSBO-TISGTR, Unit 2: STSBO-TISGTR</td>
<td>4.8E-09</td>
<td>1.4E+04</td>
<td>6.7E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR12 - Unit 1: STSBO-TISGTR, Unit 2: ISLOCA</td>
<td>3.6E-10</td>
<td>3.3E+04</td>
<td>1.2E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR13 - Unit 1: ISLOCA, Unit 2: LTSBO</td>
<td>1.8E-08</td>
<td>2.7E+04</td>
<td>4.9E-04</td>
<td>3%</td>
</tr>
<tr>
<td>PWR14 - Unit 1: ISLOCA, Unit 2: STSBO-Base</td>
<td>1.8E-09</td>
<td>2.7E+04</td>
<td>4.9E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR15 - Unit 1: ISLOCA, Unit 2: STSBO-TISGTR</td>
<td>3.6E-10</td>
<td>3.3E+04</td>
<td>1.2E-05</td>
<td>0%</td>
</tr>
<tr>
<td>PWR16 - Unit 1: ISLOCA, Unit 2: ISLOCA</td>
<td>2.7E-11</td>
<td>4.8E+04</td>
<td>1.3E-06</td>
<td>0%</td>
</tr>
<tr>
<td><strong>Total (Sum)</strong></td>
<td><strong>1.5E-05</strong></td>
<td><strong>N/A [Note 2]</strong></td>
<td><strong>4.8E-03 [Note 3]</strong></td>
<td><strong>29% [Note 5]</strong></td>
</tr>
</tbody>
</table>
Appendix IV. Risk Profiles for Both Representative Nuclear Power Plant Sites by Risk Metric and Safety Goal Policy Alternative

IV.A. Notes for Risk Profile Figures

1. A *risk profile* is a commonly-used display in NPP probabilistic risk analysis (PRA) studies for illustrating the relative contributions of different categories of accident scenarios to the total mean value of a specified risk metric. The risk profiles in this appendix are provided to facilitate identification of: (1) categories of accident scenarios that contribute significantly (≥ 10%) to each selected risk metric; (2) how the relative importance of different categories of accident scenarios differs by safety goal policy alternative—which is driven by the effect of including the risk contribution from two-unit accident scenarios; and (3) how the relative importance of different categories of accident scenarios and the effect of including two-unit accident scenarios differs by representative NPP site.

2. Each figure in this appendix includes four panels. In each figure, the top-left panel (Panel I) and top-right panel (Panel II) illustrate the risk profile for the representative boiling-water reactor (BWR) site (Peach Bottom) for the risk metric identified in the corresponding caption. In each figure, the bottom-left panel (Panel III) and bottom-right panel (Panel IV) illustrate the risk profile for the representative pressurized-water reactor (PWR) site (Surry) for the same risk metric. The panels on the left of each figure (Panel I and Panel III) respectively illustrate the risk profile for the representative BWR and PWR site under Option 1—the safety goal policy alternative that represents the status quo.
and assumes only the risk contribution from single-unit accident scenarios is included in calculating the total mean value of the identified risk metric. The panels on the right of each figure (Panel II and Panel IV) respectively illustrate the risk profile for the representative BWR and PWR site under Option 2—the safety goal policy alternative that represents a hypothetical expansion in scope and assumes the risk contributions from both single-unit and two-unit accident scenarios are included in calculating the total mean value of the identified risk metric.

3. Figures in this appendix are organized into three sections, one for each public health perspective evaluated as part of this dissertation research: (1) individual radiological health risk perspective; (2) societal radiological health risk perspective; and (3) societal non-radiological health risk perspective. Comparison of corresponding panels across figures provides insight into how the relative contributions of different categories of accident scenarios differs by risk metric.

4. Within each figure, comparison of panels in the same row (i.e., Panel I and Panel II, Panel III and Panel IV) provides insight into how, for each risk metric and NPP site, the relative contributions of different categories of accident scenarios differs by safety goal policy alternative—which is driven by the effect of including the risk contribution from two-unit accident scenarios.

5. Within each figure, comparison of panels in the same column (i.e., Panel I and Panel III, Panel II and Panel IV) provides insight into: (1) for each risk metric and safety goal policy alternative, how the relative contributions of different categories of accident scenarios differs by NPP site; and (2) for each risk metric, how the effect of including two-unit accident scenarios differs by NPP site.
IV.B. Individual Radiological Health Risk Perspective
Figure IV-1. Risk Profiles for Average Individual Early Fatality Risk (0-1 mile). Accident scenarios involving significant releases of Iodine-131 (I-131) dominate average individual early fatality risk within one mile of each representative NPP site. For the BWR site, single-unit or two-unit accident scenarios that include STSBO-Base—which involves an I-131 release fraction of 12% of reactor core inventory—comprise the entire average individual early fatality risk profile, with two-unit accident scenarios contributing 69% to the total mean value for Option 2 (Panel II). For the PWR site, single-unit or two-unit accident scenarios that include ISLOCA—which involves an I-131 release fraction of 16% of reactor core inventory—comprise the entire average individual early fatality risk profile, with two-unit accident scenarios contributing 29% to the total mean value for Option 2 (Panel IV).
Figure IV-2. Risk Profiles for Average Individual Latent Cancer Fatality Risk (0-10 miles). Panel I and Panel III respectively show that, for both the BWR and PWR sites, the categories of single-unit accident scenarios represented by the more likely LTSBO scenarios dominate Option 1 average individual latent cancer fatality risk within ten miles of the site. Panel II and Panel IV respectively show that single-unit accident scenarios and two-unit accident scenarios that include LTSBO also dominate Option 2 average individual latent cancer fatality risk for both sites, with two-unit accident scenarios contributing 23% to the total mean value for the BWR site (Panel II) and 27% to the total mean value for the PWR site (Panel IV).
IV.C. Societal Radiological Health Risk Perspective
Figure IV-3. Risk Profiles for Total Early Fatality Risk (0-50 miles). Accident scenarios involving significant releases of I-131 dominate total early fatality risk within 50 miles of each representative NPP site. For the BWR site, single-unit or two-unit accident scenarios that include STSBO-Base—which involves an I-131 release fraction of 12% of reactor core inventory—comprise the entire total early fatality risk profile, with two-unit accident scenarios contributing 64% to the total mean value for Option 2 (Panel II). For the PWR site, single-unit or two-unit accident scenarios that include ISLOCA—which involves an I-131 release fraction of 16% of reactor core inventory—comprise the entire total early fatality risk profile, with two-unit accident scenarios contributing 29% to the total mean value for Option 2 (Panel IV).
Figure IV-4. Risk Profiles for Total Latent Cancer Fatality Risk (0-50 miles). Panel I and Panel III respectively show that, for both the BWR and PWR sites, the categories of single-unit accident scenarios represented by the more likely LTSBO scenarios dominate Option 1 total latent cancer fatality risk within 50 miles of the site. Panel II and Panel IV respectively show that single-unit accident scenarios and two-unit accident scenarios that include LTSBO also dominate Option 2 total latent cancer fatality risk for both sites, with two-unit accident scenarios contributing 26% to the total mean value for the BWR site (Panel II) and 27% to the total mean value for the PWR site (Panel IV).
IV.D. Societal Non-Radiological Health Risk Perspective
Figure IV-5. Risk Profiles for Total Emergency Phase Population Relocation Risk (0-50 miles). Panel I and Panel III respectively show that, for both the BWR and PWR sites, the categories of single-unit accident scenarios represented by the more likely LTSBO scenarios dominate Option 1 total emergency phase population relocation risk within 50 miles of the site. Panel II and Panel IV respectively show that single-unit accident scenarios and two-unit accident scenarios that include LTSBO also dominate Option 2 total emergency phase population relocation risk for both sites, with two-unit accident scenarios contributing 23% to the total mean value for the BWR site (Panel II) and 17% to the total mean value for the PWR site (Panel IV).
Figure IV-6. Risk Profiles for Total Recovery Phase Population Relocation Risk (0-50 miles). Panel I and Panel III respectively show that, for both the BWR and PWR sites, the categories of single-unit accident scenarios represented by the less likely and more rapidly progressing scenarios (STSBO-Base for the BWR Site, STSBO-TISGTR and ISLOCA for the PWR site) dominate Option 1 total recovery phase population relocation risk within 50 miles of the site. Panel II and Panel IV respectively show that single-unit accident scenarios and two-unit accident scenarios that include these scenarios also dominate Option 2 total recovery phase population relocation risk for both sites, with two-unit accident scenarios contributing 22% to the total mean value for the BWR site (Panel II) and 27% to the total mean value for the PWR site (Panel IV).
Appendix V. Risk Curves for Both Representative Nuclear Power Plant (NPP) Sites by Risk Metric and Safety Goal Policy Alternative

V.A. Notes for Risk Curve Figures

1. A risk curve (also termed exceedance frequency curve) is a commonly used display in NPP probabilistic risk analysis (PRA) studies for illustrating the variability in risk results arising from random processes that influence two elements of the risk triplet for each accident scenario: (1) frequency (or probability of frequency if an integrated uncertainty analysis is performed)—which depends primarily on the combinations of events that give rise to the accident scenario; and (2) conditional consequences—which depend primarily on the potential weather conditions that can exist at the time an accident scenario occurs.\(^3\)\(^4\) The horizontal axis of a risk curve represents a range of levels for a consequence metric of interest, with units that correspond to that consequence. The vertical axis of a risk curve represents the mean frequency of exceeding a specific consequence level due to the full spectrum of potential accident scenarios that are modeled for the NPP site of interest. Risk curves are typically displayed using a logarithmic scale on both the horizontal and vertical axes (i.e., log-log scale) due to the orders of magnitude changes that are typically observed in the exceedance frequency and conditional consequences across the spectrum of accident scenarios modeled in a NPP PRA.

2. To determine the mean frequency of exceeding a specific consequence level due to potential accidents involving the NPP site of interest, perform the following
three steps in sequence: (1) locate the specific consequence level on the horizontal axis; (2) extend a vertical line to the point at which this vertical line intersects the risk curve; and (3) extend a horizontal line from this point to the point at which this horizontal line intersects the vertical axis.

3. The consequence level at which each risk curve begins on the left represents the minimum conditional consequence level observed across all weather trials for all accident scenarios included for each safety goal policy alternative that result in non-zero consequences.

4. The consequence level at which each risk curve terminates on the right represents the maximum conditional consequence level observed across all weather trials for all accident scenarios included for each safety goal policy alternative that result in non-zero consequences.

5. Each figure in this appendix includes two panels. In each figure, the top panel (Panel I) illustrates the risk curve for the representative boiling-water reactor (BWR) site (Peach Bottom) for the risk metric identified in the corresponding caption. In each figure, the bottom panel (Panel II) illustrates the risk curve for the representative pressurized-water reactor (PWR) site (Surry) for the same risk metric.

6. Figures in this appendix are organized into three sections, one for each public health perspective evaluated as part of this dissertation research: (1) individual radiological health risk perspective; (2) societal radiological health risk perspective; and (3) societal non-radiological health risk perspective. Comparison of corresponding panels across figures provides insight into how the variability in
risk results arising from the random processes described in Note 1 differs by risk metric.

7. Within each figure, comparison of Panel I and Panel II provides insight into how, for each safety goal policy alternative, the variability in risk results arising from the random processes described in Note 1 differs by NPP site.

8. Within each panel, comparison of the blue risk curve for Option 1 and the orange risk curve for Option 2 provides insight into how, for each risk metric and NPP site, the variability in risk results arising from the random processes described in Note 1 differs by safety goal policy alternative—which is driven by the effect of including the risk contribution from two-unit accident scenarios.

9. For all risk curves, across all risk metrics and both representative NPP sites, including the contribution from two-unit accident scenarios (i.e., comparing the risk curve for Option 1 to the risk curve for Option 2) has one or both of the following effects at the upper end of the consequence spectrum: (1) it shifts the risk curve up, which means that for a specified consequence level, the frequency of exceeding that consequence level increases; and/or (2) it extends the risk curve to the right, which means that the maximum consequence level observed across all accident scenarios and all weather trials increases—with lower-frequency, higher-consequence two-unit accident scenarios giving rise to the extreme right tail of the Option 2 risk curve.

10. Captions for each figure examine results for specified points of interest on the Option 1 and Option 2 risk curves. These points vary by risk metric and typically correspond to a benchmark consequence level that can be used to put the observed results into a familiar and understandable perspective.
V.B. Individual Radiological Health Risk Perspective
For the representative BWR site (Panel I), the mean frequency of exceeding an average individual early fatality risk level of 1E-06 (1 chance in 1,000,000) is 1E-07 per year for Option 1 and 2E-07 per year for Option 2. This means that an accidental release from the representative BWR site that results in an average individual early fatality risk level of greater than 1E-06 within one mile is expected to occur about once every 10 million years if only single-unit accident scenarios are considered, and once every 5 million years if both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), the mean frequency of exceeding an average individual early fatality risk level of 1E-06 is 7E-08 per year for Option 1 and 1E-07 per year for Option 2. This means that an accidental release from the representative PWR site that results in an average individual early fatality risk level of greater than 1E-06 within one mile is expected to occur about once every 14 million years if only single-unit accident scenarios are considered, and once every 10 million years if both single-unit and two-unit accident scenarios are considered. To put these results in perspective, a very large asteroid capable of causing a global catastrophe is expected to occur about once every 50 million years, with a 90% confidence interval that ranges from a frequency of once every 20 million years to one every 1 billion years.74
Figure V-2. Risk Curves for Average Individual Latent Cancer Fatality Risk (0-10 miles). For the representative BWR site (Panel I), the mean frequency of exceeding an average individual latent cancer fatality risk level of $1 \times 10^{-6}$ (1 chance in 1,000,000) is $3 \times 10^{-5}$ per year for both Option 1 and Option 2. This means that an accidental release from the representative BWR site that results in an average individual latent cancer fatality risk level of greater than $1 \times 10^{-6}$ within ten miles is expected to occur about once every 30,000 years, regardless of whether two-unit accident scenarios are considered. However, Panel I further shows that including two-unit accident scenarios causes: (1) the mean exceedance frequency to increase for risk levels at or above $1 \times 10^{-4}$ (1 in 10,000 chance); and (2) the maximum risk level observed across all weather trials and accident scenarios to increase from $5 \times 10^{-4}$ (1 in 2,000 chance) to $1 \times 10^{-3}$ (1 in 1,000 chance). For the representative PWR site (Panel II), the mean frequency of exceeding an average individual latent cancer fatality risk level of $1 \times 10^{-6}$ is $8 \times 10^{-5}$ per year for Option 1 and $9 \times 10^{-5}$ per year for Option 2. This means that an accidental release from the representative PWR site that results in an average individual latent cancer fatality risk level of greater than $1 \times 10^{-6}$ within ten miles is expected to occur about once every 13,000 years if only single-unit accident scenarios are considered, and about once every 11,000 years if both single-unit and two-unit accident scenarios are considered.
V.C. Societal Radiological Health Risk Perspective
The maximum total number of early fatality cases observed over all weather trials and all modeled accident scenarios for either option is 2E-02 for the representative BWR site (Panel I, Option 2) and 5E-01 for the representative PWR site (Panel II, Option 2). These counterintuitive non-integer results are due to population modeling features that make it possible for an accident scenario to result in more than zero but less than one early fatality. To obtain some practical insights, the mean exceedance frequency for the maximum total number of early fatality cases observed for Option 2 can be used as an upper limit on the mean frequency of accident scenarios that result in at least one early fatality for each NPP site. For the representative BWR site (Panel I), this mean exceedance frequency is 4E-12, which means that an accidental release that results in at least one early fatality among individuals residing within 50 miles is expected to occur no more than about once every 250 billion years, even if both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), this mean exceedance frequency is 2E-14, which means that an accidental release that results in at least one early fatality among individuals residing within 50 miles is expected to occur no more than about once every 50 trillion years, even if both single-unit and two-unit accident scenarios are considered.
Figure V-4. Risk Curves for Total Latent Cancer Fatality Risk (0-50 miles). For the representative BWR site (Panel I), the mean frequency of exceeding one latent cancer fatality is $3 \times 10^{-5}$ per year for both Option 1 and Option 2. This means that an accidental release from the representative BWR site that results in at least one excess latent cancer fatality among individuals residing within 50 miles is expected to occur about once every 30,000 years, regardless of whether only single-unit accident scenarios or both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), the mean frequency of exceeding one latent cancer fatality is $8 \times 10^{-5}$ per year for Option 1 and $9 \times 10^{-5}$ per year for Option 2. This means that an accidental release from the representative PWR site that results in at least one excess latent cancer fatality among individuals residing within 50 miles is expected to occur once every 12,500 years if only single-unit accident scenarios are considered, and about once every 11,000 years if both single-unit and two-unit accident scenarios are considered.
V.D. Societal Non-Radiological Health Risk Perspective
Figure V-5. Risk Curves for Total Emergency Phase Population Relocation Risk (0-50 miles). Approximately 70,000 individuals were relocated during the emergency phase of response to the 2011 Fukushima nuclear accident.\textsuperscript{8} For the representative BWR site (Panel I), the mean frequency of exceeding 70,000 relocated individuals is 1E-05 per year for Option 1 and 2E-05 per year for Option 2. This means that an accidental release from the representative BWR site that results in emergency phase relocation of more than 70,000 individuals within 50 miles is expected to occur once every 100,000 years if only single-unit accident scenarios are considered, and once every 50,000 years if both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), the mean frequency of exceeding 70,000 relocated individuals is 5E-05 per year for Option 1 and 6E-05 per year for Option 2. This means that an accidental release from the representative PWR site that results in emergency phase relocation of more than 70,000 individuals within 50 miles is expected to occur once every 20,000 years if only single-unit accident scenarios are considered, and once every 17,000 years if both single-unit and two-unit accident scenarios are considered.
Figure V-6. Risk Curves for Total Recovery Phase Population Relocation Risk (0-50 miles). Approximately 100,000 individuals were relocated during the recovery phase of response to the 2011 Fukushima nuclear accident.\textsuperscript{8} For the representative BWR site (Panel I), the mean frequency of exceeding 100,000 relocated individuals is 4E-06 per year for Option 1 and 5E-06 per year for Option 2. This means that an accidental release from the representative BWR site that results in recovery phase relocation of more than 100,000 individuals within 50 miles is expected to occur once every 250,000 years if only single-unit accident scenarios are considered, and once every 200,000 years if both single-unit and two-unit accident scenarios are considered. For the representative PWR site (Panel II), the mean frequency of exceeding 100,000 relocated individuals is 6E-09 per year for Option 1 and 8E-09 per year for Option 2. This means that an accidental release from the representative PWR site that results in recovery phase relocation of more than 100,000 individuals within 50 miles is expected to occur about once every 170 million years if only single-unit accident scenarios are considered, and once every 125 million years if both single-unit and two-unit accident scenarios are considered.
References


76. Tobin M, Kuritzky A. *Update on Staff Plans to Apply the Full-Scope Site Level 3 PRA Project Results to the NRC’s Regulatory Framework.* SECY-12-0123. Washington, DC: U.S. Nuclear Regulatory Commission; 2012.


# Curriculum Vitae

## PERSONAL INFORMATION
- **Full name:** Daniel Wayne Hudson
- **Date of birth:** August 19, 1974
- **Place of birth:** Simi Valley, CA

## EDUCATION

<table>
<thead>
<tr>
<th>Institution Name</th>
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<tr>
<td>Johns Hopkins University Bloomberg School of Public Health</td>
<td>Baltimore, MD</td>
<td>Doctor of Philosophy, Health Policy and Management</td>
<td>AUG 2017</td>
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<td></td>
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<td>Graduate Certificate, Environmental and Occupational Health</td>
<td>JUL 2013</td>
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<tr>
<td></td>
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<td>Graduate Certificate, Risk Sciences and Public Policy</td>
<td>JUN 2012</td>
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<tr>
<td>University of Maryland A. James Clark School of Engineering</td>
<td>College Park, MD</td>
<td>Master of Science, Reliability Engineering</td>
<td>AUG 2016</td>
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<tr>
<td>Johns Hopkins University School of Medicine</td>
<td>Baltimore, MD</td>
<td>Doctor of Medicine Candidate</td>
<td>2007–2008</td>
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<tr>
<td>U.S. Naval Academy</td>
<td>Annapolis, MD</td>
<td>Bachelor of Science (summa cum laude), Aerospace Engineering</td>
<td>MAY 2000</td>
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## PROFESSIONAL EXPERIENCE

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<tr>
<td>U.S. Nuclear Regulatory Commission</td>
<td>Rockville, MD</td>
<td>Reliability Engineer and Risk Analyst</td>
<td>2014–Present</td>
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<td></td>
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<td>Graduate Fellow in Probabilistic Risk Analysis</td>
<td>2011–2014</td>
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<td>Technical Assistant and Probabilistic Risk Assessment Project Manager</td>
<td>2010–2011</td>
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<td>Human Factors and Reliability Engineer</td>
<td>2009–2010</td>
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<tr>
<td>Innovative Decisions, Inc.</td>
<td>Vienna, VA</td>
<td>Principal Analyst (Part-Time)</td>
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<td>U.S. Navy</td>
<td>Multiple Locations</td>
<td>Submarine Warfare Officer and Navy Diver</td>
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## EXTRACURRICULAR, VOLUNTEER, AND OTHER LEADERSHIP EXPERIENCE

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<td>Society of Decision Professionals</td>
<td>Palo Alto, CA</td>
<td>Chairman and Membership Analytics Lead, Membership Development Council</td>
<td>2014–Present</td>
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<tr>
<td>Johns Hopkins Armstrong Institute for Patient Safety and Quality</td>
<td>Baltimore, MD</td>
<td>Volunteer, Patient Safety and Healthcare Quality Improvement Projects</td>
<td>2009–Present</td>
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<td>U.S. Nuclear Regulatory Commission</td>
<td>Rockville, MD</td>
<td>Volunteer, Regulatory Information Conference</td>
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<td>Volunteer, Combined Federal Campaign</td>
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<td>Subject Matter Expert, Safety Culture Policy Statement Public Workshop</td>
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<td>U.S. Naval Academy</td>
<td>Annapolis, MD</td>
<td>President, Tau Beta Pi Engineering Honor Society</td>
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<td>Brigade Executive Officer (Senior Leadership Position)</td>
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</table>
RESEARCH EXPERIENCE

U.S. Nuclear Regulatory Commission
Reliability Engineer and Risk Analyst
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TEACHING EXPERIENCE

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PROFESSIONAL CERTIFICATIONS AND QUALIFICATIONS

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National Board of Public Health Examiners
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Federal Acquisition Institute
Contracting Officer’s Representative, Level II
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Scuba Diving International
Advanced Open Water Diver
Stuart, FL
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U.S. Navy
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Occupational Safety and Health Officer
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Nuclear Engineer Officer
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Submarine Warfare Officer
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Navy Diver
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Anti-Terrorism Officer
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Space Systems Engineering Subspecialty Designation (5500E)
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PROFESSIONAL AFFILIATIONS

Institute for Operations Research and the Management Sciences
2011–Present

Society of Decision Professionals
2011–Present

Society for Risk Analysis
2011–Present

Society for Benefit-Cost Analysis
2016–Present
## PROFESSIONAL TRAINING COURSES

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<td>Advanced Systems Analysis Programs for Hands-on Integrated Reliability Evaluations</td>
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<td>Fire Probabilistic Risk Analysis</td>
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<td>Decisions and Risk</td>
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<td>Probability, Statistics, and Bayesian Inference for Probabilistic Risk Analysis</td>
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<td>Acquisition and Program Management Training and Certification Program</td>
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<td>Computer Programming with MATLAB</td>
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<td>Ground Radio Repair, Radio Fundamentals, and Basic Electronics Courses</td>
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PUBLICATIONS AND PRESENTATIONS

Theses and Dissertations

Articles Published in Refereed Journals

Technical Reports and White Papers

Conference Presentations

DISTINCTIONS AND AWARDS

Applied Risk Management Specialty Group Award, Society for Risk Analysis 2016
Navy and Marine Corps Commendation Medal (2) 2007, 2005
Navy and Marine Corps Achievement Medal (2) 2005, 2005
Junior Officer of the Year, Submarine Squadron EIGHT 2005
Lawrence Y. Spear Honor Graduate Award, Naval Submarine School 2003
Honor Graduate Award, Nuclear Prototype Training Unit 2002
Honor Graduate Award, Naval Nuclear Power School 2002
Secretary of the Navy Distinguished Graduate Award, U.S. Naval Academy 2000
Society of the Cincinnati Prize for Leadership, U.S. Naval Academy 2000
Admiral Dewitt C. Ramsey Prize, U.S. Naval Academy 2000
Lieutenant Clarence Lewis Tibbals Memorial Award, U.S. Naval Academy 2000