APPENDIX H
SEVERE ACCIDENT RISK ANALYSIS

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ABBREVIATIONS AND ACRONYMS

Δ	delta or incremental
$	U.S. dollars
ADAMS	Agency wide Documents Access and Management System
ANS	American Nuclear Society
AP1000	Advanced Passive 1000
APET	accident progression event tree
ASME	American Society of Mechanical Engineers
ATD	atmospheric transport and dispersion
Ba	chemical element barium
B&W	Babcock and Wilcox
BWR	boiling-water reactor
C	degrees Celsius
C_i	consequences for each potential accident i
CDET	core damage event tree
CDF	core damage frequency
Ce	chemical element cerium
CE	Combustion Engineering
CFR	Code of Federal Regulations
Ci	radiation units in Curies
CPRR	containment protection and release reduction
Cs	chemical element cesium
DF	decontamination factor
DOE	U.S. Department of Energy
DW	drywell
DWF	drywell first strategy
ELAP	extended loss of alternating current power
EPA	U.S. Environmental Protection Agency
EPRI	Electrical Power Research Institute
EPZ	emergency planning zone
ESP	early site permit
ETE	evacuation time estimate
F	degree Fahrenheit
FLEX	flexible coping strategies
FR	*Federal Register*
GE	General Electric
gpm	flow rate in gallons per minute
I	chemical element iodine
IE	initiating event
ILRT	integrated leak rate testing
IPE	individual plant examination
IPEEE	individual plant examination for external events
ISLOCA	interfacing systems loss-of-coolant accident
K	degrees Kelvin
Kg/m^3	gas density in kilograms per cubic meter
Kg/s	mass flow rate in kilograms per second
KI	chemical compound potassium iodide
La	chemical element lanthanum

LERF	large early release frequency
LMT	liner melt-through
LTSBO	long-term station blackout
LWR	light-water reactor
MACCS	MELCOR Accident Consequence Code System
MCi	radiation unit in million Curies
Mo	chemical element molybdenum
Mod	modification
MW_t	megawatt thermal
NEI	Nuclear Energy Institute
NFPA	National Fire Protection Association
NPP	nuclear power plant
NRC	U.S. Nuclear Regulatory Commission
NSSS	nuclear steam supply systems
NTTF	Near-Term Task Force
OCP	operating cycle phase
OMB	Office of Management and Budget
OP	overpressurization
P_i	probability or frequency of potential accident i
PAG	protective action guide
PRA	probabilistic risk assessment
Psi	pounds per square inch
psig	pounds per square inch gauge
PWR	pressurized-water reactor
QHO	quantitative health objective
R	risk
RC	release category
RPV	reactor pressure vessel
Ru	chemical element rubidium
RuO_2	chemical compound ruthenium oxide
Ry	reactor-year
SAMA	severe accident mitigation alternative
SAMDA	severe accident mitigation design alternative
SAPHIRE	Systems Analysis Program for Hands-on Integrated Reliability Evaluations
SAWA	severe accident water addition
SAWM	severe accident water management
SBO	station blackout
SFP	spent fuel pool
SGTR	steam generator tube rupture
SOARCA	State-of-the-Art Reactor Consequence Analyses
SPAR	Standardized Plant Analysis Risk
SRM	staff requirements memorandum
STSBO	short-term station blackout
Te	chemical element tellurium
U.S.	United States
W	rate of sensible heat
WWF	wetwell first strategy
Xe	chemical element xenon

SEVERE ACCIDENT RISK ANALYSIS

H.1 PURPOSE

The purpose of this appendix is to provide guidance and best practices for use at the U.S. Nuclear Regulatory Commission (NRC) when performing probabilistic risk assessments (PRAs) and consequence analyses as part of regulatory, backfit, and environmental analyses for nuclear power reactors.

Used in conjunction with the discussion in Section 5 of this NUREG, this appendix explains how to perform the safety goal evaluation and the valuation of the public health (accident) and economic consequences (offsite property) attributes for the purposes of cost-benefit analysis. It provides references on sources of information and an overview of the tools and methods used to estimate baselines and changes in core damage frequency (CDF), large early release frequency (LERF), public health risk, and offsite economic consequences risk. Onsite risk attributes—occupational health risk (accident) and onsite property risk—are also relevant to nuclear power reactor severe accident risk but are not within the scope of this appendix. Finally, the guidance on performing offsite consequence analyses is useful for reference when conducting the severe accident mitigation alternative (SAMA) and severe accident mitigation design alternative (SAMDA) analyses that are required under the National Environmental Policy Act (see Appendix I, "National Environmental Policy Act Cost-Benefit Analysis Guidance," to this NUREG).

This appendix does not impose new requirements, establish NRC policy, or instruct NRC staff in preparing cost estimates. Rather, it provides information on accepted state-of-practice methods for estimating the frequency and consequence components of the risk from hypothetical accidents at nuclear power plants (NPPs), for the purposes of safety goal evaluations and cost-benefit analyses for regulatory, backfitting, forward fitting, issue finality, and National Environmental Policy Act environmental review analyses.

H.2 BACKGROUND

The quantification of risks associated with postulated severe accidents is an integral part of the NRC's regulatory policy and practices. A severe accident is an accident "that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment" (NRC, 2013f; ASME/ANS, 2009). The NRC uses PRAs for the severe accident risk quantification that is needed in regulatory, backfit, and environmental analyses.

The NRC has a long history of using PRA techniques to characterize severe accident risks in support of its regulatory processes and decisions. Since the completion of the seminal Reactor Safety Study (WASH-1400, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," issued October 1975 (NRC, 1975)), PRAs have provided important, actionable safety insights through a number of different studies. In the late 1970s, the NRC used insights from PRA in consideration of topics, including the likelihood of loss-of-coolant accidents, the reliability of direct current power supplies, and the effectiveness of alternate containment designs (NRC, 2016c). In the early 1980s, the NRC relied on PRA techniques to address unresolved safety issues involving accidents such as the anticipated transient without scram (NRC, 1978) and station blackout (SBO) rules (NRC, 1988b). The NRC considered risk arguments in support of licensee requests to extend equipment outage times and the Commission used information from licensee-sponsored PRAs to inform its decision in 1985 to allow continued operation of the Indian Point power plants (NRC, 2016c).

In 1985, the Commission issued a policy statement on severe accidents, which recognized that plant-specific PRAs had exposed unique vulnerabilities to severe accidents and were a potential source of significant new safety information to identify instances of undue risk (NRC, 1985). This policy statement led to the issuance of Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities—10 CFR 50.54(f)," dated November 23, 1988 (NRC, 1988a), asking each licensee to conduct an individual plant examination (IPE) to identify plant-specific vulnerabilities to severe accidents and report the results to the Commission, and later to Generic Letter 88-20, Supplement 4, "Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities—10 CFR 50.54(f)," dated June 28, 1991 (NRC, 1991), which focused on severe accidents initiated by external events. As a result, 74 PRAs representing 106 U.S. NPPs were completed; the assessments calculated CDF and LERF[1] and gave the utilities a method for tracking improvements made in terms of risk abatement and cost effectiveness (Keller and Modarres, 2005). The NRC documents its staff summary and evaluation of licensee submittals under this program in NUREG-1560, "Individual Plant Examination Program: Perspectives on Reactor Safety and Plant Performance," issued December 1997 (NRC, 1997a), and NUREG-1742, "Perspectives Gained from the Individual Plant Examination of External Events (IPEEE) Program—Final Report," issued April 2002 (NRC, 2002), for the IPEs and IPEEEs, respectively. The NRC had also sponsored an assessment of the risks from severe accidents in five commercial nuclear power plants in the United States which was published in 1990 as NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" (NRC, 1990b). NUREG-1150 and supplementary studies based on NUREG-1150 were the main sources of information and basis for the NRC's 1997 NUREG/BR-0184, "Regulatory Analysis Technical Evaluation Handbook, Final Report" (NRC, 1997b); for example, see NUREG/BR-0184,

[1] LERF is defined as "The frequency of a rapid, unmitigated release of airborne fission products from the containment to the environment that occurs before effective implementation of offsite emergency response, and protective actions, such that there is a potential for early health effects" (NRC, 2013f).

Table 5.3 and Appendix B.4.

The Commission formally endorsed the use of PRA methods in nuclear regulatory activities in its 1995 policy statement (NRC, 1995a), which includes the following precepts:

- The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and data and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

- PRA and associated analyses (e.g., sensitivity studies, uncertainty analyses, and importance measures) should be used in regulatory matters, where practical within the bounds of the state-of-the-art, to reduce unnecessary conservatism associated with current regulatory requirements, regulatory guides, license commitments, and staff practices. Where appropriate, PRA should be used to support the proposal for additional regulatory requirements in accordance with 10 CFR 50.109 (Backfit Rule) [Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109, "Backfitting"].

- PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review.

The 1995 policy statement introduced the concept of risk-informed regulation; which solidified the role of PRA methods and results in regulatory decisionmaking. Today, the NRC conducts risk analyses for a wide range of regulatory activities and processes. Examples of activities that rely on PRA include:

- Regulatory analysis and backfit analysis: PRAs are used to determine whether additional new regulatory requirements for licensees could lead to a substantial safety improvement. Potential benefits such as reduced public health risk or reduced risk of offsite economic consequences are quantified as part of the cost-benefit analysis to justify new or amended rules or guidance.

- New reactor certification and licensing: 10 CFR 52.47, "Contents of Applications; Technical Information," requires that an application for standard design certification contain a description of the plant-specific PRA and its results. A similar requirement applies to combined license applicants in 10 CFR 52.79, "Contents of Applications; Technical Information in Final Safety Analysis Report."

- Risk-informed decisionmaking:

    - Changes in plant licensing basis: Operating reactor licensees may use risk information to support a voluntary change from a plant's current licensing basis to a new licensing basis. Regulatory Guide 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis" (current version), provides guidance on the use of PRA findings and risk insights to a support licensee request for changes to a plant's licensing basis.

- Reactor oversight: The NRC's regulatory framework for reactor oversight is risk-informed and performance based.[2] The Reactor Oversight Process uses performance indicators and inspection findings that are color coded according to safety/risk significance. Within the Reactor Oversight Process's strategic performance area of reactor safety, significance determinations of inspection findings and events rely on plant-specific risk information, such as the changes in CDF and LERF.

- Environmental reviews: The licensee prepares an environmental report and submits it to the NRC for independent evaluation as part of an application for license renewal for an existing reactor, a design certification application for a new reactor, and a construction and operating license application for a new reactor. These reports are required to include SAMA or SAMDA evaluations to identify potential features or actions that would prevent or mitigate the consequences of a severe accident. These requirements appear in 10 CFR 51.53(c)(3)(ii)(L) for operating reactor license renewal applicants; 10 CFR 51.55, "Environmental Report; Standard Design Certification," for new reactor design certification applicants; and 10 CFR 51.75, "Draft Environmental Impact Statement—Construction Permit, Early Site Permit, or Combined License," for new reactor construction permits, early site permits,[3] and combined license environmental impact statements. A PRA and offsite consequence analysis would support whether these SAMA are cost-beneficial.

In addition, the 2011 accident at the Fukushima Dai-ichi NPP in Japan initiated a large-scale effort by the staff to identify potential modifications to equipment and operational requirements to address the lessons learned from this disaster. The NRC undertook a number of major regulatory analyses to inform Commission decisions. Notable examples are listed below, with additional information available in enclosures to this appendix as indicated. The following analyses are regulatory analyses that supported these NRC decisions. The enclosures to this appendix summarize these analyses and highlight the approaches and evaluation criteria that were used, the information that was provided, the results and insights, and the resulting Commission decision, if applicable. These enclosures are intended to provide useful examples for performing these types of analyses.

- SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012 (NRC, 2012h) and SRM-SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated May 19, 2013 (NRC, 2013h). See also Enclosure H-3.

- SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling-Water Reactor Rulemaking Activities," dated June 18, 2015 (NRC, 2015a) and SRM-SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling-Water Reactor Rulemaking Activities," dated August 19, 2015 (NRC, 2015c). See also Enclosure H-4.

- The spent fuel pool (SFP) study supporting the evaluation of expedited transfer or spent fuel, SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake

[2] https://www.nrc.gov/reactors/operating/oversight/rop-description.html

[3] This applies if a design has been chosen at the early site permit stage.

Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," dated October 9, 2013 (NRC, 2013e). See also Enclosure H-5.

- COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 12, 2013 (NRC, 2013g) and SRM-COMSECY-13-0030, "Staff Requirements—Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 23, 2014 (NRC, 2014h). See also Enclosure H-6.

- Mitigation of beyond-design basis events is described in SECY-15-0065, "Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events," dated April 30, 2015 (NRC, 2015d) and SRM-SECY-15-0065, "Staff Requirements—Proposed Rulemaking: Mitigation of Beyond-Design-Basis Events," dated August 27, 2015 (NRC, 2015f).

These activities have resulted in a more consistent and technically justified application of PRA and severe accident consequence analysis in the NRC's regulatory process and serve as the basis for this guidance. The following sections explain the risk information, tools, methods, and approaches that are used to conduct these analyses.

H.3 SEVERE REACTOR ACCIDENT RISK INFORMATION USED IN SAFETY GOAL EVALUATION AND COST-BENEFIT ANALYSIS

The NRC uses a risk analysis framework to determine when a proposed requirement may meet the substantial additional protection standard and to provide some of the metrics needed to weigh the costs against the benefits of a regulatory action. Evaluating the benefits associated with a regulatory action requires the quantification of both the likelihood and the conditional consequences of fission product release for a spectrum of hypothetical severe accident scenarios. The complexity of the risk analysis depends on the type of analysis to be conducted. This appendix should be used with Section 2.1 of this NUREG to understand the level of effort needed for each type of analysis and the factors that should be used to determine which analysis is appropriate.

Staff should consult the most current PRA information available when beginning a new analysis.

A basic principle of this NUREG is that each analysis should be adequate for its intended application in terms of the type of information supplied, the level of detail provided, the level of uncertainty, and the availability of design margin. In general, the severe accident risk analysis considers plant systems and operator responses to initiating events leading to core damage (Level 1 PRA) and accident progression to the release of fission products to the environment (Level 2 PRA), while combining estimates of radiological release category frequencies and their associated consequences (Level 3 PRA) to produce risk estimates. This section details the technical approach used to complete each portion of the risk evaluation. These discussions assume familiarity with the concepts of risk as related to the nuclear industry, as well as knowledge of event- and fault-tree terminology. The analyst should consult existing PRAs and standard references[4] for further information on these concepts. Sections H.4 through H.6 provide specific guidance for performing analyses.

H.3.1 Probabilistic Risk Assessment Model Selection Guidance

The purpose of this section is to provide the analyst with guidance on selecting PRA models to perform safety goal screenings and estimate the potential public health benefits (from avoided accidents) associated with a proposed regulatory action. Performing these evaluations requires a PRA model to analyze the effects of the proposed action. The most important considerations for selecting the PRA model are its scope and its level of detail, which together should be sufficient to assess the issues of concern.

[4] For instance, NUREG/CR-2300, “PRA Procedures Guide: A Guide to the Performance of Probabilistic Risk Assessments for Nuclear Power Plants,” issued January 1983 (NRC, 1983a), and NUREG-0492, “Fault Tree Handbook,” issued January 1981 (NRC, 1981).

H.3.1.1 Probabilistic Risk Assessment Model Scope

The NPP PRA models can vary in scope, depending on their intended application or use. As summarized in Table H-1, the scope of a PRA is defined by the extent to which various options for the following five factors are modeled and analyzed:

(1) Radiological sources: The NPP sites contain multiple sources that could potentially release radioactive material into the environment under accident conditions. Although most PRA models focus on the reactor core, other important sources that could be modeled in the PRA to estimate the public health accident risk from an NPP site include (1) spent nuclear fuel (both wet and dry storage), (2) fresh nuclear fuel, and (3) radiological waste storage tanks.

(2) Exposed population: In estimating the numbers of radiological health effect cases attributable to a postulated nuclear accident, both onsite and offsite populations may be considered. Typical NPP PRA models estimate the radiological health risk to members of the general public located at various distances from the NPP site. Although these PRA models do not consider the risk to onsite workers and first responders to a nuclear accident, the radiological health risks to these groups typically are considered as part of other attributes included in a regulatory analysis (e.g., occupational health (accident)).

(3) Initiating event hazard groups: Initiating events cause the plant to deviate from its intended operating state and challenge plant control, safety systems, and operator actions designed to prevent reactor core damage and the release of radioactive material to the environment. These events include failure of equipment from (1) internal causes (e.g., transients, loss-of-coolant accidents, internal floods, internal fires) or (2) external causes (e.g., earthquakes, high winds, tsunamis). In an NPP PRA model, similar causes of initiating events are organized by hazard group and are then assessed using common assumptions, methods, and data to characterize their effects on the plant.

(4) Plant operating states: In determining the public risk from NPP operations, it is important to consider not only the response of the plant to initiating events occurring during at-power operation but also its response to initiating events occurring while the plant is in other operating states, such as low-power and shutdown. Plant operating states are used to subdivide the plant operating cycle into unique states defined by various characteristics (e.g., reactor power, coolant temperature, coolant pressure, coolant level, equipment configuration) so that the plant response can be assumed to be the same for all initiating events that occur when a plant is assumed to be in a particular plant operating state.

(5) End state (level of risk characterization): The NPP PRA models can be used to calculate risk metrics at different end states. The text below discusses in more detail the three different end states or levels of risk characterization that traditionally have been used in NPP PRA models.

Table H-1 Options Defining Scope of Commercial NPP PRAs

Factor	Scoping Options for Commercial NPP PRAs
Radiological sources	Reactor core(s) Spent nuclear fuel (SFP and dry cask storage) Other radioactive sources (e.g., fresh fuel and radiological wastes)
Exposed population	Offsite population
Initiating event hazard groups	Internal hazards • Traditional internal events (transients, loss-of-coolant accidents) • Internal floods • Internal fires
	External hazards • Seismic events (earthquakes) • Other site-specific external hazards (e.g., high winds, external flooding)
Plant operating states	At-power Low-power Shutdown
End state/Level of risk characterization	Level 1 PRA: Initiating event to onset of core damage Level 1 plus LERF: Level 1 plus limited scope Level 2, which is sufficient for the purpose of calculating LERF Level 2 PRA: Initiating event to radioactive material release from containment Level 3 PRA: Initiating event to offsite radiological consequences

The most important aspects to consider when evaluating the scope of a PRA model is to ensure that it includes significant risk contributors that are relevant to the evaluation of a proposed regulatory action and that the level of detail is appropriate with respect to scope, level of detail, and technical acceptability.

H.3.1.2 The Structure of Traditional Nuclear Power Plant Probabilistic Risk Assessment Models

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. Three sequential adverse end states that can occur in the progression of postulated NPP accident scenarios define these levels (1) onset of damage to the nuclear fuel in the reactor core (termed 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 referred to as “offsite radiological consequences”).

Figure H-1 illustrates the overall logic and structure of traditional NPP PRA models, including the types of results that are produced at each level.

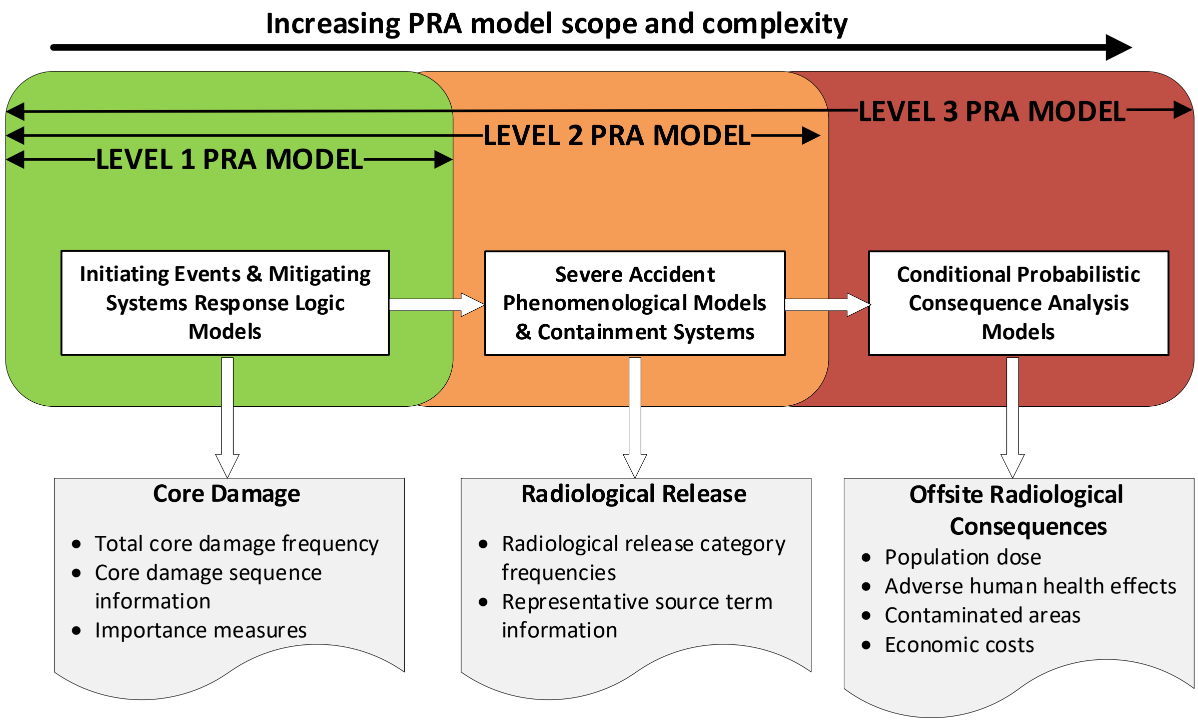

Figure H-1 Overall Logic and Structure of Traditional NPP PRA Models

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 consequence results for each release category. For each outcome of interest, the consequences are then summed across all radiological release categories to estimate the mean annual risk of that outcome.

The first step in conducting the analysis is to identify the potential source of risk (e.g., reactor core, spent fuel, dry cask storage), reactor operating states (e.g., at-power, low-power, shutdown), and hazards of concern (e.g., internal events, external events, all hazards) for analysis. The potential source of risk will usually be determined by the objective statement described in Sections 2.3.1 and 2.3.2 of this NUREG, which provide guidance on defining the regulatory problem statement and identifying regulatory alternatives. A complete assessment of alternatives that includes all relevant accident scenarios may require the development of plant-specific, full-scope Level 3 PRAs for each plant type of interest. However, this may exceed the required level of detail necessary for a regulatory analysis. For most regulatory analyses, the regulatory problem statement will delineate the accident initiators and sequences to be considered.

H.3.2 Risk Metrics for Evaluating Substantial Safety Benefit

For potential backfit considerations, it is useful to have an approximation of the range of the CDFs and LERFs for relevant classes of plants. Section 2.4.1 of this NUREG describes the quantitative risk thresholds for substantial safety benefit. The NRC uses LERF instead of the historical conditional containment failure probability (see for example, Regulatory Guide 1.174). The analyst has access to a current body of CDF and LERF information of operating NPPs from a variety of sources. These sources include the NRC's plant-specific Standardized Plant Analysis Risk (SPAR) models, risk information in SAMA analyses supporting license renewal applications, and license amendment requests supporting risk-informed regulatory applications such as those for risk-informed in-service inspection (NRC, 2003).

Figures H-2 (CDF) and H-3 (LERF) show representative distributions of point estimates for CDF and LERF, published in NUREG-2201, “Probabilistic Risk Assessment and Regulatory Decisionmaking: Some Frequently Asked Questions,” issued September 2016 (NRC, 2016c). The purpose of these figures in this appendix is to provide a general illustration of the distribution of CDFs and LERFs. These figures depict the CDF and LERF for a subset of the U.S. fleet of operating power reactors, based on information readily available through NRC regulatory applications. As noted in NUREG-2201, the CDFs are based on 2016 estimates for 61 units from license amendment requests to change requirements or SAMA analyses as part of the environmental evaluation conducted by license renewal applicants. The earliest result is from a 2002 analysis, but over 80 percent of the results are from 2008 or later. The estimates are based on PRAs with different scopes, for example, the majority included internal plus external event initiators while a minority included internal event initiators only.

The point estimate for CDFs range from about 4×10^{-6} per reactor-year to approximately 1×10^{-4} per reactor-year, with a mean and median of about 5×10^{-5} per reactor-year. The point estimates for LERFs range from about 8×10^{-8} per reactor-year to approximately 3×10^{-5} per reactor-year, with a mean of approximately 4×10^{-6} per reactor-year and a median of about 3×10^{-6} per reactor-year. The source information for these estimates typically do not include uncertainty estimates. NUREG-2201 also notes that it is important to recognize:

> - [P]ast PRAs have consistently shown that potential vulnerabilities (and therefore plant risk) are highly plant specific.
> - Design and operational changes addressing lessons identified by PRAs can lead to significant changes in CDF...
> - The above estimates for total CDF are developed by adding the CDFs estimated for different accident scenarios.
> - The CDF contributions from accidents caused by internal hazards (e.g., floods, fires) and external events (e.g., earthquakes, high winds, and external floods) can be significant.
>
> (Source: NUREG-2201, p. 36)

It is important to note that external events are sometimes out-of-scope or handled much less rigorously than internal events (for example, in SAMA analyses for operating reactors). See additional discussion in Section H.4.2, “Sources of Information,” and table notes under Tables H-3 and H-4. Similar information is available for new and advanced reactors (see Section H.5.2), with the exception that large release frequency is used instead of LERF.

As noted above, the analyst should access available risk information that is current at the time of a future regulatory or cost-benefit analysis. Figures H-2 and H-3 provide an example based on 2016 data for a subset of operating reactor units, with the aforementioned limitations.

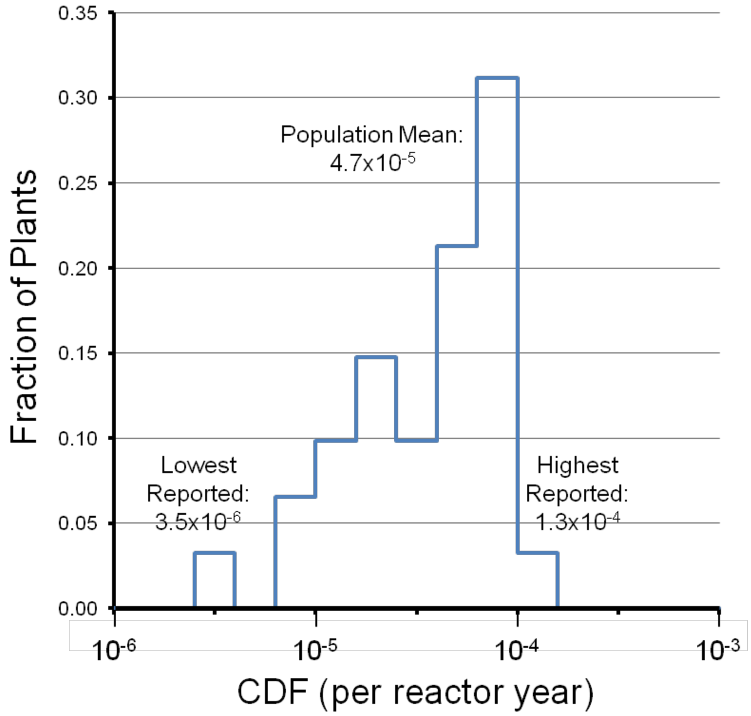

Figure H-2 Distribution of 2016 Point Estimates for Total CDF, U.S. Plants
(Source: NUREG-2201, Figure 4-3)

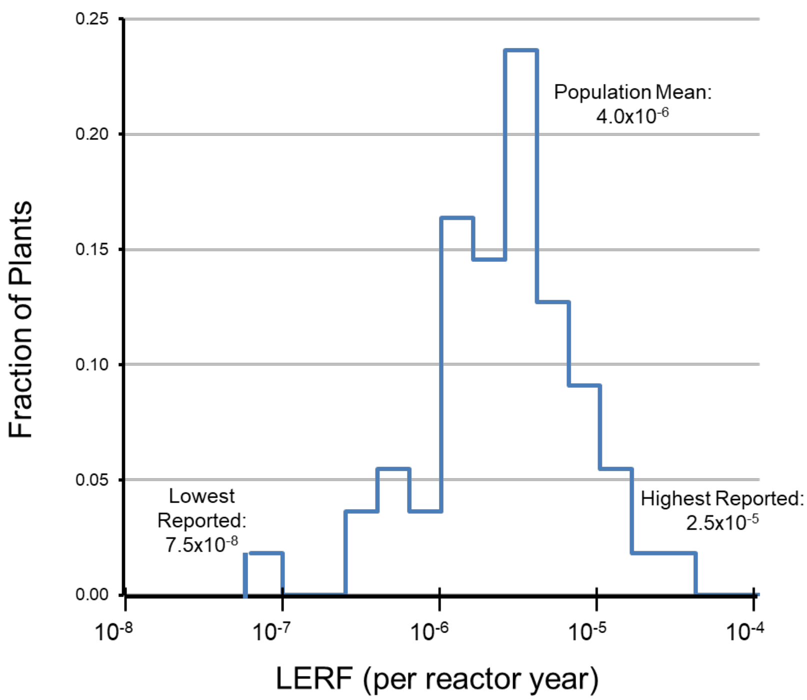

Figure H-3 Distribution of 2016 Point Estimates for LERF, U.S. Plants
(Source: NUREG-2201, Figure 5-2)

H.3.3 Common Analysis Elements

Risk (R) is, summed over the spectrum of potential accidents, the product of (1) the probability (or frequency) (P_i) and (2) associated consequences (C_i) for each potential accident (i) in the spectrum, as shown in the equation below:

$$R = \sum_i P_i C_i$$

Hence, estimating the public health (accident) risk and offsite economic consequences (offsite property damage) risk in a cost-benefit analysis for a proposed action requires the estimation of both (1) the change in probabilities (frequencies) and (2) the change in consequences associated with accidents in the spectrum of relevant accidents. Therefore, the common analysis elements include the following:

- An accident sequence analysis to identify the relevant accidents

- Quantification of frequencies associated with individual accident sequences for the probability/frequency portion of the risk equation

- Quantification of the public health and offsite economic consequence associated with each accident sequence, for the consequence portion of the risk equation

The following sections discuss these elements in greater detail.

H.3.3.1 Accident Sequence Analysis

An accident sequence analysis systematically identifies risk-significant accident sequences and quantifies their frequency. Logic models provide the probabilistic framework for assessing the change in risk associated with a regulatory analysis alternative. These models consist of event trees to identify the set of possible accident sequences that lead to fission product release and rely on accident progression simulations performed for a specific accident sequence to understand how a combination of successes and failures affects the facility. The following examples are for a nuclear power plant, but the principles apply to all NRC-regulated facilities.

PRA Logic Model Structure

One PRA modeling approach is to construct logic models using event trees and fault trees. An event tree represents different plant and operator responses in terms of sequences of undesired system states, such as core damage or fission product release, that could occur following an initiating event. The probabilistic (Level 1 and Level 2 frequency) portions of an accident sequence analysis are assessed using Core Damage Event Trees (CDETs) and Accident Progression Event Trees (APETs). A fault tree identifies different combinations of basic events (e.g., initiating events; failures of systems, structures, and components; and human failure events) that could lead to a system failure. Fault tree models are linked to the event tree sequences and allow for the identification and evaluation of minimal cut sets—the minimum combinations of events needed to result in an adverse end state of interest (e.g., core damage). When linked together, these logic structures provide an integrated perspective that can capture major system dependencies.

Care should be taken to ensure that the modeling is sufficiently detailed and is technically adequate to provide the needed confidence in the results—for its use in the regulatory analysis and for its role in the integrated decision process, which is critical for coherent decision-making. Because the standards and industry PRA programs are not prescriptive, there is some freedom on how to model these logic structures. The choice of specific assumptions, a particular approximation, or a modeling choice or simplification may, however, influence the results. These underlying assumptions and approximations made in the development of the PRA model give rise to uncertainty and should be explicitly identified and quantified to aid the decisionmaker in understanding the results and the potential range of costs and benefits. The treatment of uncertainty and sensitivity analysis are further discussed in Section H.6.

PRA Logic Model Level of Detail

Much like the scope, the level of detail of an NPP PRA model can vary, depending on its intended application or use. The level of detail is defined by the degree to which (1) the actual plant is modeled and (2) the unlimited range of potential accident scenarios is simplified. Although the goal of a PRA is to represent the NPP as-designed, as-built, and as-operated as realistically as practicable, some compromises are made to keep the PRA model manageable, considering time and resource constraints.

For each of the technical elements that comprise a PRA model, the level of detail may vary by the extent to which the following is true:

- Plant systems and operator actions are credited in modeling plant-specific design and operation

- Plant-specific operating experience and data for the plant’s structures, systems, and components are incorporated into the model

- Realism (as opposed to intentional conservatism) is incorporated into analyses that predict the expected plant and operator responses

Furthermore, the logic structures (e.g., event trees and fault trees) in the model are simplified representations of the complete range of potential accident scenarios. Simplifications are made through underlying assumptions and approximations such as (1) the consolidation into representative hazard groups of initiating event causes and (2) the screening out of certain equipment failure modes.

Although the level of detail needed for an NPP PRA model is largely dependent upon the requirements associated with its intended use (e.g., a PRA should meet the relevant American Society of Mechanical Engineers [ASME] and American Nuclear Society [ANS] PRA standards for operating reactor licensing changes), at a minimum, it needs to be detailed enough to model the major system dependencies and to capture the significant risk contributors.

The level of effort required to construct these logic models depends upon the availability of information and preexisting models developed for the specific site of interest and on the amount of information that is obtainable from the licensee. The NRC has developed SPAR models for all NPPs used to support various risk-informed activities. However, depending upon the scope of the regulatory analysis, these models may need to be expanded to address other hazards or plant conditions. To the extent possible, the analyst should use existing information, in addition to related research efforts,[5] to complete the regulatory analysis efficiently. Qualitative insights may be needed to supplement incomplete quantitative modeling.

Assumptions about which systems will be available (or should be probabilistically considered) are dependent upon the type of initiating event being considered. For example, if the initiating event is seismically induced, consideration should be given to whether a given safety system realistically would be available. The assumptions used in developing the event trees should be clearly delineated for the systems that are probabilistically considered. In constructing the event trees, systems or modes of operations for which reliability data are not available should not be credited or probabilistically considered. The analyst should document for reference these assumptions and all hardware-related failure event probabilities that are incorporated in the CDETs and APETs.

H.3.3.2 Quantification of Change in Accident Frequency

The change in accident frequency is a key factor for several of the cost-benefit analysis attributes. Estimates of the change in accident frequencies resulting from a proposed NRC action are based on the effects of the action on appropriate parameters in the accident equation. Examples of these parameters might be system or component failure probabilities, including those for the facility's containment structure. The estimation process involves two steps—(1) identification of the parameters affected by a proposed NRC action, and (2) estimation of the values of these affected parameters before and after the action takes place.

The parameter values are substituted in the accident equation to yield the base- and adjusted-case accident sequence frequencies. The sum of their differences is the reduction in accident frequency caused by the proposed NRC action. The frequency of accident sequence *i* initiated by event *j* is

$$F_{ij} = \sum_k M_{ijk}$$

where M_{ijk} = the frequency, *F*, of minimal cut set *k* for accident sequence *i* initiated by event *j*
Source: (NRC, 1997b).

[5] For example, related research efforts include SPAR external events modeling (https://saphire.inl.gov/current_models_public.cfm), fire risk research under National Fire Protection Association (NFPA) 805, "Performance-Based Standard for Fire Protection for Light-Water Reactor Electric Generating Plants" (current version) (https://www.nrc.gov/reactors/operating/ops-experience/fire-protection/protection-rule/protection-rule-overview.html), and generic issue evaluations (https://www.nrc.gov/about-nrc/regulatory/gen-issues/dashboard.html).

A minimal cut set represents a unique and minimum combination of occurrences at lower levels in a structural hierarchy (e.g., component failures that are typically represented by basic events in PRA model fault trees) needed to produce an overall occurrence (e.g., facility damage) at a higher level. It takes the form of a product of these lower level occurrences. The affected parameters comprise one or more of the multiplicative terms in the minimal cut sets. Thus, the change in accident sequence frequency *i*, between the base model and the adjusted model that incorporates the proposed action, is

$$\Delta F_{ij} = \left[(F_{ij})_{base} - (F_{ij})_{adjusted}\right] = \sum \left[(M_{ijk})_{base} - (M_{ijk})_{adjusted}\right]$$

Source: (NRC, 1997b)

The changes in accident frequency for each affected accident sequence are added. Reduction in accident frequency is algebraically positive; increase is negative. This equation assumes that the model structure remains valid for risk evaluations after a proposed action. It is possible for a proposed action to result in a change to the model structure (e.g., by adding or removing top events in an event tree). Therefore, in addition to potentially changing the values of parameters that comprise a base-case set of minimal cut sets, a proposed action can change the structure of the minimal cut sets and create new minimal cut sets that were not included in the base case. This would require an evaluation beyond quantification of the above equation, which only quantifies the change of frequencies of existing minimal cut sets.

Each accident sequence that ends in core damage is binned for further analysis into a plant damage state with other core damage sequences having plant conditions that are expected to result in similar accident progression behavior. The frequencies of the sequences with a core damage end state are summed to estimate the CDF for an initiating event. The characteristics that define each plant damage state bin comprise the initial conditions for the APET. Similarly, the APETs evaluate the containment response to those sequences that result in core damage and provide the frequencies of sequences with end states of release to the environment.

Source terms are binned into release categories based on release characteristics such as magnitude and timing of release. Binning both the plant damage states, and source terms reduces the total number of accident progression and consequence simulations that are required. In summing the CDF and LERF/large release frequency, the analyst should consider all significant accident sequences. Significant accident sequences, as defined in Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities” (current version), are those that, when ranked, compose 95 percent of the CDF or LERF, or that individually contribute more than 1 percent to the CDF.

In practice, the computation of change in the frequency of CDF and release categories for both the standard analysis and the major effort uses PRA software, such as Systems Analysis Program for Hands-On Integrated Reliability Evaluations (SAPHIRE), are discussed in Enclosure H-1, “Description of Analytical Tools and Capabilities,” to this appendix.

H.3.3.3 Quantification of Change in Consequences

Many analyses may assume that new consequence evaluations will not be needed. If the change in risk can be captured through a change in accident sequence frequencies only, then the overall risk equation can use the existing public health and economic consequence assessments associated with those accident sequences. This assumption is embedded when existing population dose and offsite economic consequence multipliers (e.g., "population dose factors" in Section 5.3.2.1 of this NUREG) are used for severe accident sequences. However, if a proposed action affects an accident's conditional consequences, then the risk quantification approach should explicitly account for the change in conditional consequences, as noted at the end of Section 5.3.2.1.1 of this NUREG. If the existing PRA model does not adequately capture the change in risk associated with the proposed change, then the PRA model should be revised to support the analysis.

Regulatory analyses involving large light-water reactors historically have been estimated using a 50-mile radius from the site (see Section 5.2.1 of this NUREG). The analyst chooses the distance based on the potentially affected area (e.g., where offsite population dose and offsite property damage is incurred). Offsite consequences for other distances[6] have been considered in recent detailed analyses where individual plants with site-specific information were evaluated. Section H.5 and Enclosures H-4 through H-6 to this appendix discuss examples. For small modular reactors and advanced reactors, the radius should be chosen based on design-specific details, site characteristics, and precedents.

H.3.3.4 Identification and Estimation of Affected Parameters

An action may affect accident frequencies only, accident consequences only, or both accident frequencies and consequences. Actions that may change existing PRA model structures (e.g., by adding or removing events in an event tree or changing consequences of existing accident sequences) will require additional analysis steps compared to actions that affect only the relative frequencies of existing accident sequences and associated consequences.

If appropriate PRA models are available, these can be used to identify the affected parameters. For example, all NPP PRA studies include accident sequences involving loss of emergency alternating current power. If the minimal cut sets used in the analytical modeling of these sequences contain parameters appropriate to an action related to loss of emergency alternating current power, then these PRA studies would be appropriate for use in the analysis. In this case, the analyst can readily identify the affected parameters and their estimated values.

Within the scope of an analysis, the identification of affected parameters may require more than the direct use of existing PRA models. Existing studies may need to be modified. The effort may involve (1) performing an expanded or independent analysis of the accident sequences associated with an action, using previous studies only as a guideline, or (2) combining several existing PRA studies to form a composite study more applicable to a generic action. Care should be taken to ensure that assumptions, modeling, and uncertainty characterization are appropriate and valid to support decisionmaking.

Assuming the analyst has identified affected parameters, the next step is to estimate the base- and adjusted-case values of the affected parameters, which are then used to estimate the base- and adjusted-case total accident sequence frequencies and associated consequences. The sum of the differences between the base- and adjusted-cases is the change in accident frequency, the consequence resulting from the action, or both.

In some cases, additional modeling is required, where identification of affected parameters

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[6] The analyst should also consider the capabilities and range of validity of analytical tools when selecting these distances.

requires the type of analysis associated with a much greater level of detail and, most likely, a significantly expanded scope. NRC programs related to unresolved generic safety issues for power reactors offer examples of where major efforts were required in the past. Such programs tend to be multiyear tasks. The expected level of detail and quality of analysis should be consistent with current standard practice and may entail peer review.

H.4 GRADED APPROACH TO ANALYSIS

As with most areas of NRC's regulation (e.g., NRC's "Strategic Plan: Fiscal Years 2018-2022" [NRC, 2018a]), staff are expected to take a risk-informed approach to severe accident risk analyses supporting regulatory analyses. NRC's Office of Nuclear Reactor Regulation Office Instruction LIC-504, "Integrated Risk-Informed Decision-Making Process for Emergent Issues" (NRC, 2014i) describes different levels of approach, namely a graded approach, to using risk information that, while tailored to decision-making for emergent issues, is conceptually appropriate to the use of risk information in regulatory analyses too. A graded approach is one where the level of rigor applied depends on the importance, e.g., risk significance and applicability (see for example, discussion in Management Directive 6.4, "Generic Issues Program" [NRC, 2015g]). As noted in LIC-504, Regulatory Guide 1.174, and NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making" (NRC, 2017a), it is particularly important to assess uncertainties in the risk analyses and understand how uncertainties may affect the comparison of risk measures with decision criteria.

In some cases, an initial screening-type analysis may be sufficient to disposition the evaluation of a potential regulatory action. For example, if it is necessary to show a substantial safety benefit and possibly to get an initial assessment of whether a potential regulatory change may be cost-beneficial, existing compilations of risk information may be sufficient to make the determination (this would be analogous to answering "yes" to the question in NRC's LIC-504 Section 4.2.2, "Is the Issue Clearly of Low Safety Significance?" [NRC.2014i]). For such an approach, the potential benefits should be maximized, and (if pursuing an initial cost-benefit assessment) the potential costs minimized, to ensure that a potentially warranted action is not unduly screened out. Furthermore, uncertainty in these screening or bounding-type analyses and its potential impact should be considered.

In the absence of a new major-effort analysis, existing risk information would be used, e.g., by selecting the maximum CDF for the class of affected plants and the highest known conditional consequences within the class of affected plants. Current CDFs at the time of an analysis are available, such as in the information sets used to create Figures H-2 and H-3 above. While the conditional consequences may be harder to find, several sources of information (discussed in Section H.5.2) exist and could provide the needed estimates. The highest conditional consequences for a class of plants typically will be tied to the highest population sites. Both 10-mile- and 50-mile-radius populations should be considered for large light-water reactors; for small modular reactors and advanced reactors, the radius could be chosen based on design-specific details and precedence (such as EPZ and Protective Action Guides [PAGs]). The joint consideration of a site's meteorological profile, population distribution, and licensed thermal power (since total radiological releases for a given accident are expected to scale with core power) is important. The offsite populations residing within 50 miles of the operating NPPs in the United States varied from 180,000 to 17 million, according to the 2000 and 2010 censuses (NRC, 1996 and supplements). As of 2019, the licensed thermal power for individual large light-water reactors in the United States varied from 1,700 megawatts thermal (MW_t) to 4,400 MW_t (NRC, 2019b).

As discussed in Section 5.3.2 of this NUREG, the estimation of the avoided public health effects and avoided offsite economic consequences is calculated from current risk information from existing studies. The avoided consequences are computed by multiplying the change in frequency of each significant release category by its consequence metrics and then applying a summation over all affected release categories. This approach should only be used if the staff deems that existing risk studies adequately capture the accident scenarios, associated frequencies and consequences, for the issue under consideration.

At the simplest level, the analysis assumes values of affected parameters are readily available or can be derived easily. Sources of data that are readily accessible include existing PRA studies, which provide parameter values in forms appropriate for accident frequency calculations (e.g., frequencies for initiators and unavailability or demand failure probabilities for subsequent failures of systems, structures, and components).

After identifying base- and adjusted-case values for the parameters in the plant-risk equation that are affected by the proposed regulatory action (see Section 5.3.2 of this NUREG), the analyst calculates the change in accident frequency as the sum of the differences between the base- and adjusted-case values for all affected accident sequences.

Uncertainties are prevalent in any risk assessment and should be addressed (see Section H.6.3.1 for a discussion on different kinds of uncertainties). For example, an error factor on the best estimate of the reduction in total accident frequency may be used to estimate high and low values for the sensitivity calculations in the analysis for power reactor facilities. Past analyses have used error factors of 5-10 or more, depending on the events analyzed[7]. Error factors from the specific risk assessment being used, if available, or knowledge of typical error factors from current analogous risk assessments, should be employed.

An analyst who is unable to identify affected parameters for an action can estimate changes in accident frequency using professional judgment. Expert opinion also plays a prime role in estimating adjusted-case parameter values. Typically, existing data are applied to yield base-case values, leaving only engineering judgment for arriving at adjusted-case values. Reaching consensus among multiple experts can increase confidence, and the magnitudes of parameter values normally encountered in PRA studies can serve as rough guidelines.

At a more detailed level, but still within the scope of a standard analysis, the analyst may conduct reasonably detailed statistical modeling or extensive data compilation when values of affected parameters are not readily available. While existing PRA studies may provide some data for use in statistical modeling, the level of detail required normally would be greater than they could provide. Statistical modeling may use stochastic simulation methods and involve statistical analysis techniques using extensive data.

NUREG/CR-2800, "Guidelines for Nuclear Power Plant Safety Issue Prioritization Information Development," issued September 1983 (NRC, 1983b), discusses the calculation of change in core melt accident frequency for power reactors, and provides examples. Such calculations are typical for a standard cost-benefit analysis. A useful reference is Nuclear Energy Institute (NEI)-05-01, Revision A, "Severe Accident Mitigation Alternatives (SAMA) Analysis Guidance Document," issued November 2005 (NEI, 2005), because SAMA analyses follow a similar process to that of regulatory and cost-benefit analyses. A SAMA analysis includes searches for potential generic industry and plant-specific improvements to address important risk contributors, and cost-benefit analyses to evaluate these potential improvements.

[7] See for example: https://nrcoe.inl.gov/resultsdb/publicdocs/AvgPerf/ComponentUR2015.pdf

H.4.1. Example of Approach

The staff analysis summarized in Enclosure H-3, "Summary of Detailed Analyses for SECY-12-0157, 'Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments,'" provides an example of a practical modern approach to what was historically called a "standard" analysis. To evaluate the potential risk reduction benefit of the proposed action, the staff first reviewed insights from available risk studies. These sources of risk information included (1) the IPEs completed in response to Generic Letter 88-20 (NRC, 1988a, NRC, 1997a), (2) applicable risk-informed license amendment requests, which in this case were the requests for integrated leak rate testing (ILRT) (see Table 2 of NRC, 2012h), and (3) SAMA analyses submitted with license renewal applications for operating NPPs (NRC, 1996, and supplements). The ILRT license amendment requests were considered because they estimated post-core-damage containment-related risk benefits that informed the evaluation of potential benefits of installing containment venting systems. The staff collected the following information from these sources:

- Identification of the conditional containment failure probabilities from the class of plants under consideration (e.g., boiling-water reactors [BWRs] with Mark I and Mark II containments), for base-case conditions in the IPEs and ILRTs, as well as sensitivity to extended ILRT intervals

- Identification of dominant contributors to early containment failure

- Evaluation of whether past SAMA analyses considered filtered severe accident venting, and if so, whether they found it to be a potentially cost-beneficial plant improvement at the time of the license renewal application

This evaluation of available risk insights contributed to the technical approach for evaluating potential benefits by helping the staff to develop the branches on the event tree for sequence evaluation and benefit quantification (see Enclosure H-3 to this NUREG for more details of this analysis).

A safety goal evaluation is required as part of a regulatory analysis in which regulatory alternatives are analyzed to determine whether they are generic safety enhancement backfits subject to the substantial additional protection standard. To perform the safety goal evaluation, the staff should analyze the regulatory alternatives to directly compare the potential safety benefits to the QHOs for average individual early fatality risk and average individual latent cancer fatality risk described in the Commission's Safety Goal Policy Statement[8] (NRC, 1986). To determine the relative costs and benefits, the analyst should compare each of the alternatives to the regulatory baseline.

[8] In 1986, the NRC published the Safety Goal Policy Statement, whose objective was to, "establish goals that broadly define acceptable level of radiological risk" to the public from nuclear power plant operation (NRC, 1986). This policy stated two qualitative safety goals, supported by two quantitative objectives which are commonly called QHOs: (1) the risk to an average individual in the vicinity (1 mile) 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; and (2) the risk to the population in the area (within 10 miles) 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. Since the QHOs are tied to the prompt fatality risks and cancer fatality risks from all other causes in the U.S., the actual QHOs can change over time.

A successful strategy used in the past for the safety goal evaluation is to employ a high-level and conservatively high estimate to maximize the potential benefit of a regulatory alternative for comparison to the regulatory baseline, to determine whether an alternative may meet the substantial safety benefit threshold. For example, in the Containment Protection and Release Reduction (CPRR) regulatory analysis described in Enclosure H-4 to this appendix, the staff performed a screening analysis for the average individual latent cancer fatality risk QHO for the relevant plants—all U.S. BWRs with Mark I containments (a total of 22 units at 15 sites) and Mark II containments (a total of eight units at five sites). For this screening analysis, the staff developed a conservatively high estimate of the frequency-weighted average of an individual latent cancer fatality risk within 10 miles of the plant using the following parameter values:

- An extended loss of alternating current power (ELAP)[9] frequency value of 7×10^{-5} per reactor-year—which represented the highest value among all BWRs with Mark I and Mark II containments
- A success probability for flexible coping strategies (FLEX) equipment of 0.6 per demand—which assumed the implementation of FLEX will successfully mitigate an accident involving an ELAP 6 out of 10 times
- A conditional average individual latent cancer fatality risk of 2×10^{-3} per event—which represented the highest value among all BWRs with Mark I and Mark II containments from the detailed analyses

These assumed parameter values resulted in a conservatively high estimate of a frequency-weighted individual latent cancer fatality risk within 10 miles of approximately 7×10^{-8} per reactor-year (labelled as "High-Level Conservative Estimate" in Figure H-4), which is greater than an order of magnitude less than the QHO for an average individual latent cancer fatality risk of approximately 2×10^{-6} per reactor-year. This conservatively high estimate did not take credit for any of the accident strategies and capabilities described in the 20 CPRR alternatives and subalternatives. Figure H-4 shows the incremental benefit (in terms of individual latent cancer fatality risk on the y-axis) for each alternative on the x-axis–subalternatives within Alternatives 2 to 4 compared to the status quo, Alternative 1.

Because the conditional early fatality risk was essentially zero, a comparable analysis for the early fatality QHO was not needed.

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[9] An ELAP is defined as an SBO that lasts longer than the SBO coping duration specified in 10 CFR 50.63, "Loss of all alternating current power."

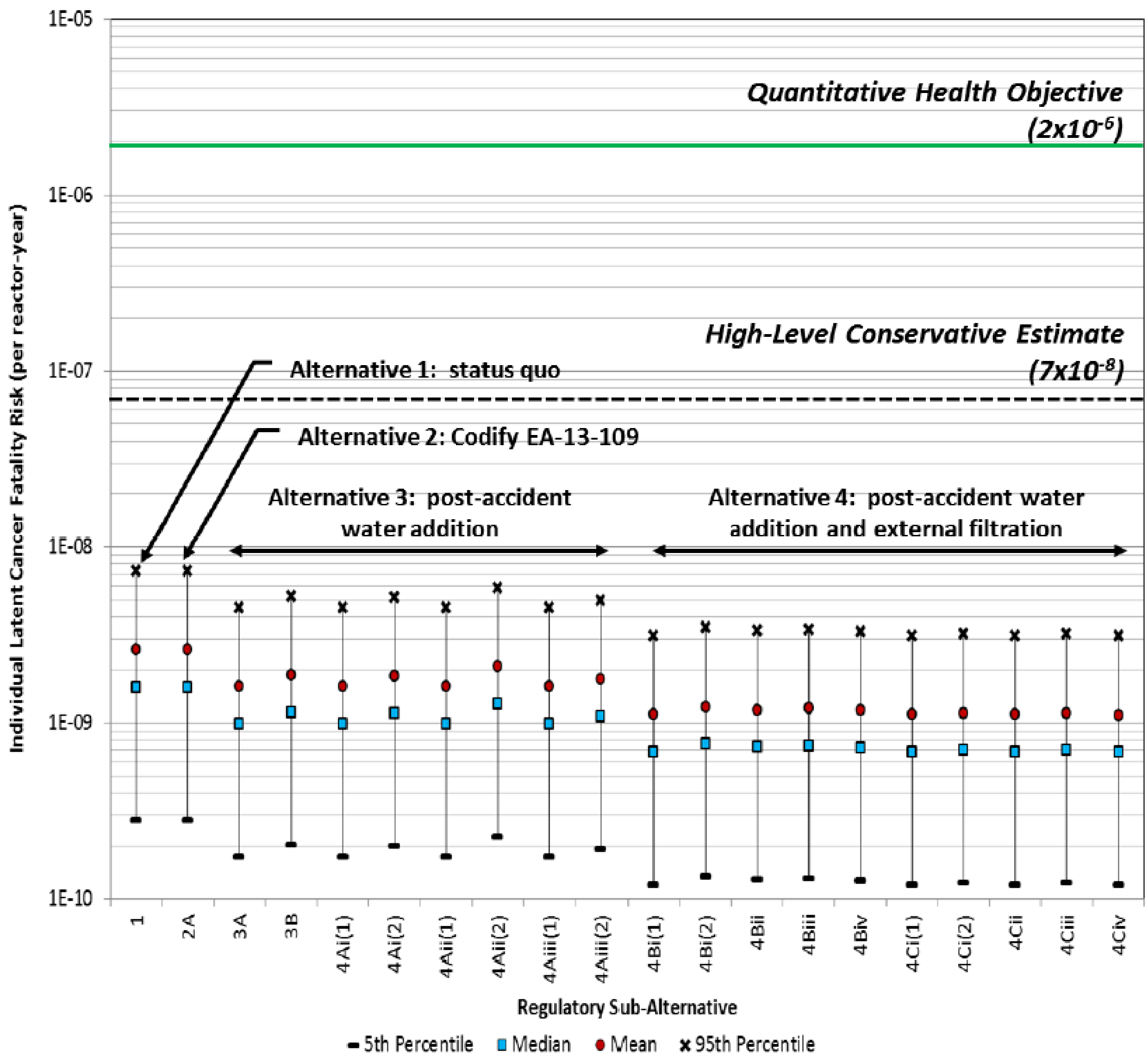


Figure H-4 Uncertainty in Average Individual Latent Cancer Fatality Risk (0–10 miles) in the 2015 Containment Protection and Release Reduction Regulatory Analysis

(Source: SECY-15-0085, Enclosure, Figure 3-3)

H.4.2. Sources of Information

As noted in the Background section above, historically, NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued December 1990 (NRC, 1990b), and supplementary studies based on NUREG-1150, were the main sources of information for the NRC's typical regulatory analyses. The analyst should consult the SPAR Program owner to collect the most current risk information and insights at the time of a new analysis. For example, the NRC maintains SPAR models for use in the Reactor Oversight Process and other risk-informed regulatory activities, as noted in Section H.3.3.1 and discussed further in Enclosure H-1. Risk-informed applications and SAMA analyses are other examples of sources of information, as discussed further below.

Risk-informed license amendment requests[10] cover a range of plant and risk scenarios that should be consulted according to the risk scope under consideration. The 10 CFR 50.54(f) letter responses are another source of information for a variety of plant and accident types. For example, in response to the lessons learned from the Fukushima Dai-ichi accident, the NRC issued a 10 CFR 50.54(f) letter (NRC, 2012i) to all operating NPP licensees to reevaluate the seismic and flooding hazards at their sites using updated seismic and flood hazard information and present-day regulatory guidance and methodologies and, if necessary, to request that they perform a risk evaluation. The responses to the letter provide post-2012 seismic CDF and seismic LERF information for operating NPPs.[11]

The SAMA analyses may provide useful information since SAMA analyses (1) cover all nuclear steam supply systems (NSSS) and containment types for the operating fleet of NPPs (see Table H-2), as well as new reactors under construction (e.g., SAMA and SAMDA analyses for the advanced passive 1000 [AP1000]), and (2) have been evaluated for the known risk profile (e.g., different accident initiators and scenarios) for each subject plant at the time of analysis. The SAMA analyses report on the rank of contributors to CDF (see the example in Table H-3), the rank of contributors to LERF (occasionally), the rank of contributors of different release categories or containment release modes to population dose (see example in Table H-4), and the “maximum attainable benefit” in terms of the offsite dose and offsite economic cost risks (within a 50-mile radius from the plant) that would be saved if all potential accidents could be eliminated at the plant. These analyses[12] are documented in license applications and in the staff’s environmental evaluations.[13] As noted in the main body Section 2, the SAMA analyses documented in the NUREG-1437 supplements report quantitative internal events CDFs, and external events multipliers in the range of 1.2 to 12, with an average value of 3.2 (based on 51 of the 57 supplements published between 1999 and 2016 that reported external events multipliers for 82 individual reactors). This means that the total CDF was estimated to be 1.2 to 12 times the internal events CDF, with an average value of 3.2 times the internal events CDF. Additional SAMA analyses have been performed for design certifications and combined license new reactor reviews.[14] When using data from SAMA analyses, the analyst should be aware that the agency undertakes SAMA analyses to meet NEPA’s “hard look” requirement; as a result, some aspects of SAMA analyses may require further consideration before the agency relies on them to meet its obligations under the Atomic Energy Act of 1954, as amended.

Table H-2 Reactors with Published SAMA Analyses

Containment Type	NSSS Type	Plant Name	Licensed Thermal Power (MWt)	NUREG-1437[a, b] Supplement Number (unless noted otherwise)
Dry, Ambient	B&W Lowered Loop	Arkansas 1	2568	3
		Oconee 1	2568	2
		Oconee 2	2568	2
		Oconee 3	2568	2
	B&W Raised Loop	Davis-Besse	2817	52
	CE	Arkansas 2	3026	19

[10] For example, see risk-informed technical specification changes discussed here: https://www.nrc.gov/reactors/operating/licensing/techspecs/risk-management-tech-specifications.html
[11] https://www.nrc.gov/reactors/operating/ops-experience/japan-dashboard/seismic-reevaluations.html
[12] https://www.nrc.gov/reactors/operating/licensing/renewal/applications.html contains links to all NPP license renewal applications and the NRC’s reviews.
[13] https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/
[14] https://www.nrc.gov/reactors/new-reactors.html

Containment Type	NSSS Type	Plant Name	Licensed Thermal Power (MWt)	NUREG-1437[a, b] Supplement Number (unless noted otherwise)
		Calvert Cliffs 1	2737	1
		Calvert Cliffs 2	2737	1
		Millstone 2	2700	22
		Palisades	2565	27
		Saint Lucie 1	3020	11
		Saint Lucie 2	3020	11
		Waterford 3	3716	59
Large Dry, Ambient	CE 80	Palo Verde 1	3990	43
		Palo Verde 2	3990	43
		Palo Verde 3	3990	43
Mark I	GE 2	Nine Mile Point 1	1850	24
	GE 3	Dresden 2	2957	17
		Dresden 3	2957	17
		Monticello	2004	26
		Quad Cities 1	2957	16
		Quad Cities 2	2957	16
	GE 4	Browns Ferry 1	3952	21
		Browns Ferry 2	3952	21
		Browns Ferry 3	3952	21
		Brunswick 1	2923	25
		Brunswick 2	2923	25
		Cooper	2419	41
		Duane Arnold	1912	42
		Fermi 2	3486	56
		FitzPatrick	2536	31
		Hatch 1	2804	4
		Hatch 2	2804	4
		Hope Creek 1	2902	45
		Peach Bottom 2	4016	10
		Peach Bottom 3	4016	10
Mark II	GE 4	Limerick 1	3515	49
		Limerick 2	3515	49
		Susquehanna 1	3952	35
		Susquehanna 2	3952	35
	GE 5	Columbia	3544	47
		La Salle 1	3546	57
		La Salle 2	3546	57
		Nine Mile Point 2	3988	24
Mark III	GE 6	Grand Gulf 1	4408	50
		River Bend 1	3091	58
Dry, Ambient	Westinghouse 2-loop	Ginna	1775	14
		Point Beach 1	1800	23

Containment Type	NSSS Type	Plant Name	Licensed Thermal Power (MWt)	NUREG-1437[a, b] Supplement Number (unless noted otherwise)
		Point Beach 2	1800	23
		Prairie Island 1	1677	39
		Prairie Island 2	1677	39
Dry, Subatmospheric	Westinghouse 3-loop	Beaver Valley 1	2900	36
		Beaver Valley 2	2900	36
		North Anna 1	2940	7
		North Anna 2	2940	7
		Surry 1	2587	6
		Surry 2	2587	6
Dry, Ambient	Westinghouse 3-loop	Farley 1	2775	18
		Farley 2	2775	18
		Harris 1	2948	33
		Robinson 2	2339	13
		Summer	2900	15
		Turkey Point 3	2644	5
		Turkey Point 4	2644	5
Dry, Ambient	Westinghouse 4-Loop	Braidwood 1	3645	55
		Braidwood 2	3645	55
		Byron 1	3645	54
		Byron 2	3645	54
		Callaway	3565	51
		Indian Point 2	3216	38
		Indian Point 3	3216	38
		Millstone 3	3650	22
		Salem 1	3459	45
		Salem 2	3459	45
		Seabrook 1	3648	46
		South Texas 1	3853	48
		South Texas 2	3853	48
		Vogtle 1	3626	34
		Vogtle 2	3626	34
		Wolf Creek 1	3565	32
Ice Condenser	Westinghouse 4-Loop	Catawba 1	3469	9
		Catawba 2	3411	9
		D.C. Cook 1	3304	20
		D.C. Cook 2	3468	20
		McGuire 1	3411	8
		McGuire 2	3411	8
		Sequoyah 1	3455	53
		Sequoyah 2	3455	53
		Watts Bar 2	3411	NUREG-0498, Supp. 2[c]

Containment Type	NSSS Type	Plant Name	Licensed Thermal Power (MWt)	NUREG-1437[a, b] Supplement Number (unless noted otherwise)
AP1000	Westinghouse 2-Loop	Vogtle 3[d]		NUREG-1872[d]
		Vogtle 4[d]		NUREG-1872[d]

[a] Information current as of 2019
[b] NUREG-1437 and supplements are available at: https://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1437/
[c] NRC, 2013i.
[d] Under construction; NUREG-1872, "Final Environmental Impact Statement for an Early Site Permit (ESP) at the Vogtle ESP Electric Generating Plant Site," issued August 2008 (NRC, 2008)."

Table H-3 Salem Nuclear Generating Station Core Damage Frequency for Internal Events at Power

Initiating Event	CDF[1] (per year)	% Contribution to CDF[2]
Loss of Control Area Ventilation	1.8 x 10^{-5}	37
Loss of Offsite Power (LOOP)	8.1 x 10^{-6}	17
Loss of Service Water	6.6 x 10^{-6}	14
Internal Floods	4.5 x 10^{-6}	9
Transients	4.0 x 10^{-6}	8
Steam Generator Tube Rupture (SGTR)	2.7 x 10^{-6}	6
Loss of Component Cooling Water (CCW)	1.0 x 10^{-6}	2
Anticipated Transient Without Scram (ATWS)	7.4 x 10^{-7}	2
Loss of 125V DC Bus A	6.9 x 10^{-7}	2
Others (less than 1 percent each)[3]	1.8 x 10^{-6}	4
Total CDF (internal events at power)[4]	**4.8 x 10^{-5}**	**100**

[1] Calculated from Fussel-Vesely risk reduction worth (RRW) provided in response to NRC staff RAI 1.e (PSEG, 2010a).
[2] Based on internal events CDF contribution and total internal events CDF.
[3] CDF value derived as the difference between the total Internal Events CDF and the sum of the individual internal events CDFs calculated from RRW.
[4] The results only covers a fraction of the total plant risk profile, so their usefulness for regulatory decision-making may be limited for situations where the analysis is evaluating changes involving not at power or external events.
(Source: NUREG-1437, Supplement 45, Table F-1)

Table H-4 Salem Nuclear Generating Station Breakdown of Population Dose by Containment Release Mode

Containment Release Mode	Population Dose (Person-Rem[1] Per Year)	Percent Contribution[2]
Containment overpressure (Late)	42.9	55
Steam generator rupture	31.9	41
Containment isolation failure	2.3	3
Containment intact	0.2	<1
Interfacing system Loss-of-Coolant Accident (LOCA)	0.6	<1
Catastrophic isolation failure	0.4	<1
Basemat melt-through (late)	Negligible	Negligible
Total[3,4]	**78.2**	**100**

[1] One person-rem = 0.01 person-Sv
[2] Derived from Table E.3-7 of the ER (PSEG 2009).
[3] Column totals may be different due to rounding.
[4] The results only covers a fraction of the total plant risk profile, so their usefulness for regulatory decision making may be limited for situations where the analysis is evaluating changes involving not at power or external events.
(Source: NUREG-1437, Supplement 45, Table F-2)

The State-of-the-Art Reactor Consequence Analyses (SOARCA), (see Enclosure H-2 to this appendix) is another source of information for potential offsite public health consequences within the scope of the severe accident scenarios studied for three operating reactor types in the United States.[15] SOARCA analyses, including uncertainty analyses, were conducted for short-term and long-term SBO accidents at a BWR with a Mark I containment in Pennsylvania; a three-loop Westinghouse NSSS pressurized-water reactor (PWR) with a subatmospheric large, dry containment in Virginia; and a four-loop Westinghouse NSSS PWR with an ice condenser containment in Tennessee. Deterministic analyses were also conducted for an interfacing systems loss-of-coolant accident at the PWR with a large, dry containment. Consequence results were reported as individual latent cancer risks and individual early fatality risks for different radial rings out to 50 miles from the site. The SOARCA studies focused on accident progression, source term, and conditional consequences should the postulated accidents occur. The project did not include within its scope new work to calculate the frequencies associated with the postulated severe accidents. And just like information from modern plant-specific risk-informed license amendment requests, or plant-specific SAMA analyses, the SOARCA studies were conducted for specific reactor types and sites.

[15] The SOARCA analyses are documented in a series of NUREG and NUREG/CR reports (NRC, 2012a; NRC, 2012j; NRC, 2013a; NRC, 2013b; NRC, 2014a; NRC, 2014b; NRC, 2016b; NRC, 2019a; NRC, 2020).

H.5 MAJOR-EFFORT ANALYSIS

When additional rigor is required, a “major-effort” analysis is performed. Enclosures H-4 through H-6 to this appendix summarize the major-effort regulatory analyses that the staff completed in the 2013 to 2015 timeframe. This section summarizes approaches and considerations for the common technical elements in a major-effort regulatory analysis: accident sequence analysis, accident progression (Level 2 PRA) analysis, and offsite consequence (Level 3 PRA) analysis.

H.5.1 Accident Sequence Analysis

A major-effort analysis should begin with an accident sequence analysis. The analyst should consider the following factors during the development of the technical approach for selecting the relevant set of accident sequences:

- The risk evaluation should provide risk metrics for all regulatory analysis subalternatives and do so according to the approved scope, schedule, and allocated resources.

- Consistent with the NRC’s regulatory analysis guidelines, the risk evaluation should provide fleet-average risk estimates. Therefore, the technical approach should consider the impacts of plant-to-plant variability (for example, see Section H.6.2.2).

- The staff should leverage existing relevant sources of accident sequence information and develop new information where required.

- The analyst should develop CDETs to (1) model the impact of equipment failures and operator actions occurring before core damage that affect severe accident progression and the probability that regulatory alternatives are successfully implemented, (2) match the initial and boundary conditions used in the thermal-hydraulic simulation of severe accidents in MELCOR, and (3) consider mitigating strategies for beyond-design-basis external events, as applicable.

- The analyst should develop APETs to model regulatory alternatives.

Enclosures H-3 through H-6 to this appendix include discussions of the accident sequence analyses for three detailed regulatory analyses. As discussed in Enclosure H-4 to this appendix, analysts successfully used a modular approach to develop the CDETs and APETs, as shown in Figure H-5. This modeling approach streamlined the development of risk estimates for the CPRR technical basis rulemaking and provides a good example for future detailed analyses. Enclosure H-1 to this appendix describes the NRC-sponsored software, SAPHIRE. SAPHIRE can be used for accident sequence modeling with CDETs and APETs and frequency analysis.

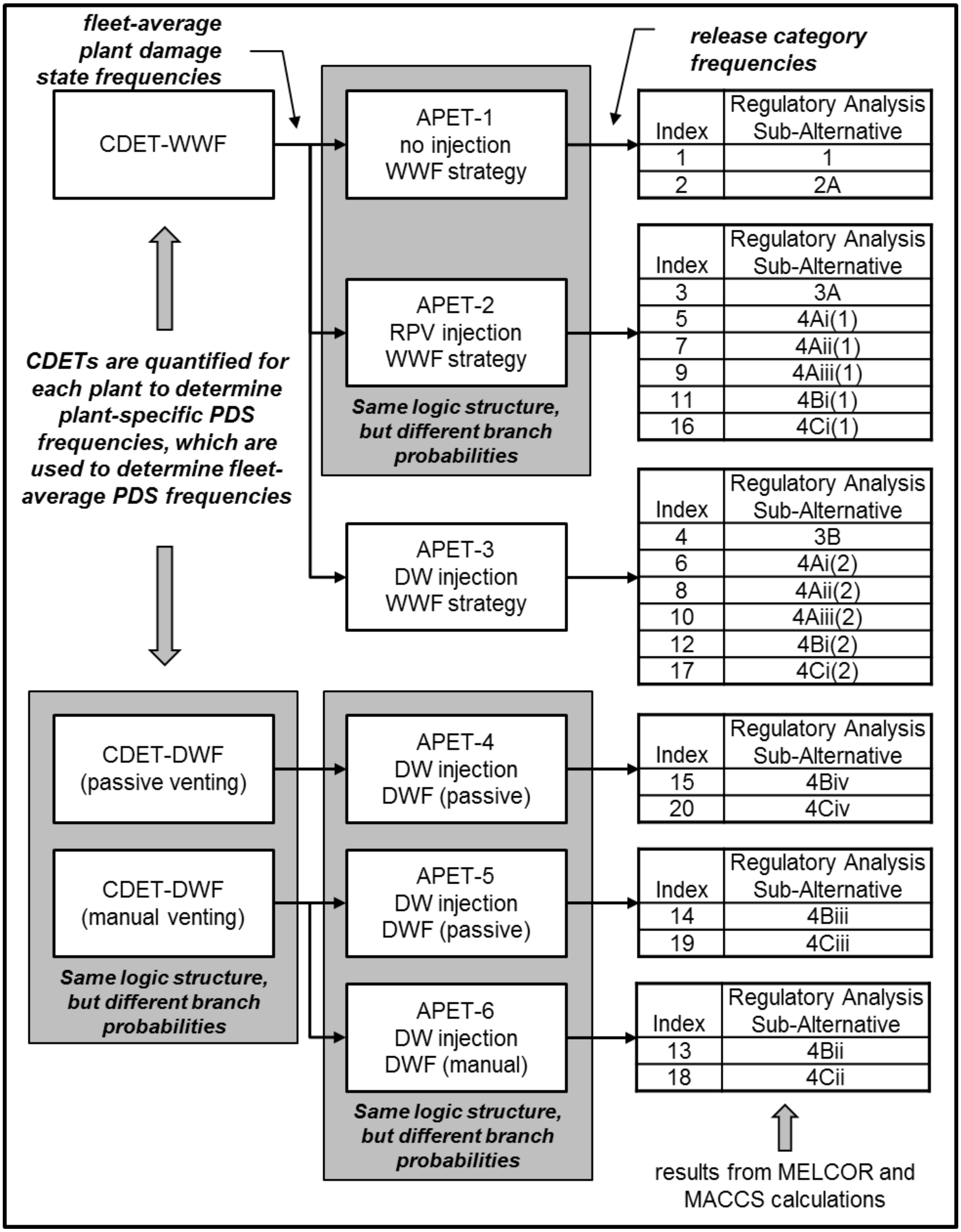


Figure H-5 Modular Approach to Event Tree Development in CPRR Analysis
(Source: NUREG-2206, issued March 2018, Figure 2-1)

H.5.2 Severe Accident Progression Analysis

The next step of a major-effort analysis is to complete a severe accident progression and source term analysis, analogous to a Level 2 PRA. The objective of the severe accident progression analysis is to generate a technical basis quantitatively characterizing thermal and mechanical challenges to engineered barriers to fission product release to the environment. This analysis provides a chronology of postulated accidents resulting in significant damage to reactor fuel and generates quantitative estimates of a radioactive material release to the environment. The staff has used the MELCOR code[16] (Humphries et al., 2015), described in Enclosure H-1 to this appendix, to model accident progression and fission product release estimates for each of the selected accident scenarios in the detailed analyses.

The two broad purposes for conducting MELCOR calculations are: (1) to evaluate reactor systems and containment thermal-hydraulics under severe accident conditions, and (2) to assess the timing and magnitude of fission products released to the environment. Three outputs—the containment temperature and pressure signatures, along with hydrogen distribution through the containment and reactor building—provide information to assess the status of reactor plant and containment integrity under varying postulated conditions. This information may provide the basis for investigating other regulatory subalternatives. Analysts use the timing and magnitude of fission product release information to characterize the source terms in the consequence analysis described in Section H.5.3.

The MELCOR calculations are deterministic in nature and simulate different possible outcomes or plant damage states, given the initial conditions that are specified in the accident sequence analysis. The analyst should choose representative plant models based on the requirements of the regulatory analysis (e.g., reflective of the relevant class(es) of NSSSs, containments, and operational features). For efficiency, the representative MELCOR plant models can use existing input decks developed for recent studies when available and relevant. For example, the regulatory analyses discussed in Enclosures H-3 and H-4 to this appendix started with the SOARCA Peach Bottom Atomic Power Station input deck for Mark I containments.

H.5.2.1 Sources of Information

NUREG/CR-7008, “MELCOR Best Practices as Applied in the SOARCA Project,” issued August 2014 (NRC, 2014a), describes the best practices in modeling approach and parameter selections that support the best estimate analyses in the 2012 SOARCA project, for a General Electric BWR with a Mark I containment and a Westinghouse 3-loop PWR with a large, dry, subatmospheric containment. The input models should follow the guidance of NUREG/CR-7008, supplemented with updates and insights from the most recent MELCOR analyses available (e.g., later SOARCA studies, such as NUREG/CR-7245, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses, issued 2019” (NRC, 2019a), for a Westinghouse 4-loop PWR with an ice condenser containment, and NUREG/CR-7155, “State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” issued May 2016 (NRC, 2016b), and future studies, such as the NRC’s Site Level 3 PRA,[17] for a Westinghouse 4-loop PWR with a large, dry containment).

[16] http://melcor.sandia.gov/
[17] https://www.nrc.gov/about-nrc/regulatory/research/level3-pra-project.html

Each operating NPP has an updated final safety analysis report that describes the facility's design bases and technical specifications and provides a safety analysis of each plant system (10 CFR 50.34(b)). The updated final safety analysis report describes plant components and containment features. The analyst can use this information to construct the MELCOR model.

IPEs provide information on the types of accidents that have a potential for occurring and the location of failures. As previously discussed, each operating plant has one of these risk analyses for internal events and many have IPEEEs.

Severe accident management guidelines are a source of information for characterizing operator and plant response to severe accidents. These guidelines are developed by the utility and provide guidance for operator actions in the event of a severe accident. These guidelines contain strategies to stop or slow the progression of fuel damage, maintain containment, and mitigate radiological releases.

H.5.2.2 MELCOR Modeling Approach

An accident progression analysis should be a collection of simulations of specific accident sequences that is conducted to understand how a regulatory alternative affects the plant and estimate the fission product release (source term) resulting from the accident sequence.

A MELCOR calculation matrix is developed to delineate runs evaluating each regulatory analysis alternative, the various potential plant lineups, and the sensitivity analyses performed for pre- and post-core damage mitigation measures. The calculations should clearly state the initial and boundary conditions for the analysis and base the model nodalization on the specific events that are being examined. The calculations should line up with APET and CDET sequences in the accident sequence analysis.

Each accident sequence is binned into a release category that is represented by a MELCOR source term. MelMACCS, which provides an interface between MELCOR and MACCS, can read a MELCOR source term and provide the following data for each source term:

- Time-dependent release fraction of chemical groups[18]
- Time-independent distribution by particle size diameter for 10 aerosol size bins characterized by geometric mean diameters
- Height of each MELCOR release pathway
- Time-dependent data needed to estimate buoyant plume rise, including rate of release of sensible heat (W), mass flow (kg/s), and gas density (kg/m^3)

The MELCOR source terms become input for the next step of the analysis, which are used to estimate the offsite consequences using the MELCOR Accident Consequence Code System (MACCS) suite of codes.

[18] For example, Noble Gases (Xe), Alkali Metals (Cs), Alkali Earths (Ba), Halogens (I), Chalcogens (Te), Platinoids (Ru), Early Transition Elements (Mo), Tetravalents (Ce), and Trivalents (La)) for each MELCOR release pathway

H.5.3 Offsite Consequence Analysis

Similar to the MELCOR analysis, the consequences discussed here are conditional and do not factor in the probability of release. The MACCS suite of codes[19] is the NRC's code system for performing offsite consequence analyses for severe accident risk assessments. The NRC uses MACCS to analyze hypothetical accident scenarios, and almost all parameters in the code may be modified. This functionality provides substantial flexibility and allows for the characterization of uncertainties. Enclosure H-1 to this appendix provides more details on the MACCS code and its capabilities.

H.5.3.1 Sources of Information

Similar to the MELCOR SOARCA best practices, NUREG/CR-7009, "MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," issued August 2014 (NRC, 2014b), describes the parameter selections that supported the best-estimate MACCS analyses in the 2012 SOARCA study. The MACCS input models should follow the guidance of NUREG/CR-7009, supplemented with updates and insights from the most recent MACCS analyses (e.g., later SOARCA studies, such as NUREG/CR-7245 and NUREG/CR-7155) and guidance. NUREG/CR-4551, Volume 2, Revision 1, Part 7, "Evaluation of Severe Accident Risks: Quantification of Major Input Parameters: MACCS Input," issued December 1990 (NRC, 1990c), describes the development of shielding parameters for NUREG-1150 is in greater detail.

H.5.3.2 MACCS Modeling Approach

There is considerable variation in site characteristics, such as population size and distribution, land use, economic values, weather, and emergency response characteristics (e.g., road networks, use of potassium iodide). Site-specific models historically have been developed for plant and containment types and then adapted using a series of sensitivity calculations to assess the potential impact of the site-specific parameters on the results. For efficiency, the analyst can use existing MACCS input decks developed for recent studies when available and relevant. For example, the regulatory analyses discussed in Enclosures H-3 and H-4 to this appendix started with the SOARCA Peach Bottom MACCS input deck.

Source Term Characterization

The source terms developed from the severe accident progression analysis with similar release fractions and release timing characteristics may be binned to reduce the number of MACCS cases that must be run. The binning should be based, at a minimum, on cumulative cesium and iodine release fractions and the warning times associated with each source term. Historically, the cesium group has been the most important for long-term offsite consequences (e.g., latent cancer fatality risk), and the iodine group has been the most important for early offsite consequences (e.g., early fatality risk).

The MelMACCS code[20] in the MACCS suite of codes is used to create a MACCS input file to represent the radiological source term developed using MELCOR. MelMACCS allows the user

[19] https://maccs.sandia.gov/
[20] https://maccs.sandia.gov/melmaccs.aspx

to associate the MELCOR mass values with an ORIGEN output to convert masses of chemical classes to activities of individual radionuclides. In addition, the code needs the following data to characterize each source term:

- Radionuclide releases are divided into hourly segments to be consistent with the hourly meteorological observations. If meteorological sampling is being used, the most risk-significant plume should be identified to align the release with the weather data for each weather bin. This is often taken to be the plume segment with the highest iodine chemical group release fraction.

- Building height and width are used to estimate the initial horizontal and vertical plume dispersion caused by building wake effects.

- Ground height in the MELCOR reference frame is used to adjust the MELCOR release heights relative to grade.

- Reference time, which is the difference between accident initiation time in MELCOR and scram time. This value, which is used to properly account for decay and ingrowth of radioactivity within MACCS, is usually zero but may be non-zero for some MELCOR simulations.

Site and Meteorological Data

MACCS uses a polar grid to model the exposures to people, land contamination, and protective actions of people and land. MACCS allows the user to choose 16, 32, 48, or 64 angular sectors for grid division. The analyst should choose 64 angular sectors to provide the greatest resolution. MACCS allows the user to divide the grid into a maximum of 35 radial rings, at specified radii from the plant. The boundaries are selected to be consistent with certain areas of interest. For example, for large LWR accidents, a radial boundary should be set at roughly 1 mile from the approximated site boundary to evaluate individual early fatalities for which the NRC's early fatality QHO applies (NRC, 1986). This boundary is set at 10 miles to approximate the plume exposure EPZ and latent fatality QHO, and at 50 miles to capture the majority of radiological and economic consequences.

The SecPop preprocessor code in the MACCS suite of codes is typically used to generate site-specific population and the economic data required for consequence calculations. Population data should be scaled forward to the year of interest from the year of the census data contained in SecPop using population growth data from the U.S. Census Bureau. Additionally, the economic values contained in SecPop are from the U.S. Department of Agriculture and U.S. Department of Commerce and should be scaled forward from the base year data to the year of interest, using the consumer price index for all urban consumers.

The analyst should obtain raw weather data for the representative site from the site meteorological towers for at least 2 full calendar years. Even though only 1 year of weather data is necessary to complete the calculation, multiple years are beneficial for comparison to ensure that the year selected is not anomalous (e.g., an abnormally dry or rainy year). The inherent assumption in using historical data to quantify the consequences of a future event is that future weather data will be statistically similar to historical data. The most complete year of data should be chosen, and any missing data filled in by NRC meteorologists in accordance with the U.S. Environmental Protection Agency's (EPA's) EPA-454/R-99-005, "Meteorological

Monitoring Guidance for Regulatory Applications," issued February 2000 (EPA, 2000). The methodology described in NUREG-0917, "Nuclear Regulatory Commission Staff Computer Programs for Use with Meteorological Data," issued July 1982 (NRC, 1982b), is used to perform quality assurance evaluations of all meteorological data. In accordance with Regulatory Guide 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants" (current version), the completeness of the recorded data (the data recovery rate) should be greater than 90 percent for the wind speed, wind direction, and atmospheric stability parameters. The nonuniform bin sampling approach may be used to capture the effects of variable weather, consistent with modeling best practices and recent consequence analyses.

Protective Action Modeling

EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," issued January 2017, describes the emergency phase 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" (EPA, 2017). Offsite response organization emergency plans are required to include detailed evacuation plans for the plume exposure EPZ (NRC and FEMA, 1980). Site-specific information should be obtained from offsite response organization emergency response plans and the licensee's evacuation time estimate (ETE) reports to support the development of timelines for protective action implementation. The protective action modeling assumptions have an important impact on offsite consequences.

MACCS input parameters related to evacuation modeling are taken primarily from the site-specific ETE reports, which the licensee develops and updates under 10 CFR 50.47 (b)(10). ETEs provide the time required to evacuate various sectors and distances within the EPZ for transient and permanent residents, and these times are used to develop response timing and travel speeds for evacuating cohorts[21] in MACCS.

Important information in an ETE report includes demographic and response data for four population segments, which may be readily converted into cohorts, if appropriate. These population segments are (1) permanent residents and transient population, (2) transit-dependent permanent residents (e.g., people who do not have access to a vehicle or are dependent upon help from outside the home to evacuate), (3) special facility residents (e.g., people in nursing homes, assisted living centers, hospitals, jails, prisons), and (4) schools, including all public and private educational facilities within the EPZ. In general, delineating the population into more cohorts (beyond these four) allows greater fidelity in modeling the emergency response of the public. In recent practice, the staff has further divided the ETE cohorts into additional groups (e.g., in order to separate the 10 percent of the permanent general population who may evacuate later than the other 90 percent of the general population).

The licensee's ETE report typically includes about 10 scenarios that vary by season, day of the week, time of day, and weather conditions, as well as other EPZ-specific situations such as special events. The ETEs do not consider most external events and their impact on road infrastructure, and it is important for the analyst to account for these impacts in the model. The Sequoyah SOARCA analysis provides an example of how the impact of seismic events may be considered in MACCS modeling (NRC, 2019a), if seismic events are important for the scope of accidents under consideration.

[21] As explained in more detail in Enclosure H-1 to this appendix, a "cohort" in MACCS is a group that is modeled as behaving similarly (e.g., evacuating at the same time and speed).

In modeling the early phase relocation actions, the dose criteria to trigger the actions should be consistent with the current EPA PAGs. In MACCS, emergency phase relocation is modeled with two user-specified dose criteria to trigger the action and a relocation time for the population affected by each dose. This modeling should consider site-specific features such as source term, site information, and local demographics.

Although decisions about cleanup and reoccupation of affected areas would involve both radiological and non-radiological considerations, it is customary in MACCS to use the dose criteria for intermediate phase relocation as a surrogate for decisions about long-term habitability. In determining the relocation and habitability dose criteria for the intermediate and long-term phases, state-specific guidance for relocation following the early phase (as a surrogate for decisions regarding habitability) should be followed when available. Absent state-specific guidance, the analyst should use the EPA relocation PAGs.

H.6 SUPPLEMENTAL ANALYSES

Much like other parts of the regulatory analysis, the extent of supplemental analyses should be commensurate with the complexity of the problem and associated uncertainties. At a minimum, the analyst should identify important sources of uncertainty and influential assumptions and evaluate their impacts on analysis outcomes. The results of these investigations should be summarized in the report provided to decision makers, as discussed in Section 7.4, Risk Integration Results and Key Insights.

<u>H.6.1 Uncertainty Analyses</u>

Appendix C, "Treatment of Uncertainty," to this NUREG contains a general discussion of uncertainties. The discussion below focuses on PRA uncertainties relevant to major-effort analyses.

H.6.1.1 Uncertainties in PRA Models

When using PRA results as part of any regulatory decisionmaking process, it is important to understand the types, sources, and potential impact of uncertainties associated with PRA models and how to treat them in the decisionmaking process. Using PRA for regulatory decisionmaking requires that the associated uncertainties and their implications be characterized. For a major-effort analysis, the models and available information for projecting severe accident consequences contain large uncertainties. The explicit identification and quantification of sources of uncertainty of a consequence analysis are necessary to aid the decisionmaker in understanding the results and the potential range of costs and benefits.

Although PRA models have several different sources of uncertainty, there are two principal categories: aleatory and epistemic. Aleatory uncertainty arises from the random nature of the basic events and phenomena (e.g., weather) modeled in PRAs. Because PRAs use probabilistic distributions to estimate the frequencies or probabilities of these basic events, the PRA model itself is an explicit model of the aleatory uncertainty. Similarly, the explicit modeling of different weather conditions in the Level 3 portion of a PRA is a treatment of aleatory uncertainty.

Epistemic uncertainties arise from incompleteness in the collective state of knowledge about how to represent plant behavior in PRA models. These uncertainties relate to how well the PRA model reflects the as-designed, as-built, as-operated plant and to how well it predicts the response of the plant to various scenarios. Since these uncertainties can have a significant impact on the interpretation and use of PRA results, it is important that they be appropriately identified and characterized and that the analysis address important uncertainties. The following three types of epistemic uncertainty are associated with PRA models:

- <u>Parameter Uncertainty</u>: Parameter uncertainty relates to the uncertainty of input parameter values. Probability distributions for the input parameters quantify the frequencies or probabilities of basic events in the PRA logic model. Importantly, this assumes that the selection of the probability distribution to model the likelihood of the basic event is agreed upon; if uncertainty exists about this selection, it is more appropriately considered model uncertainty.

- Model Uncertainty: Model uncertainty arises from a lack of knowledge of physical phenomena; failure modes related to the behavior of systems, structures, and components under various conditions; or other phenomena modeled in a PRA (e.g., the location and habits of members of the public in different exposure scenarios). This can result in the use of different approaches to modeling certain aspects of the plant and public response that can significantly impact the overall PRA model. Since uncertainty exists about which approach is most appropriate, this leads to uncertainty in the PRA results. Model uncertainty can also arise from uncertainty in the logic structure of the PRA model or in the selection of the probability distribution used to model the likelihood of the basic events in the PRA model. Sensitivity analyses typically address model uncertainties to determine the sensitivity of the PRA results to alternative modeling approaches. The ASME/ANS PRA standards (ASME/ANS, 2009, 2014, 2017) treat Level 2 and Level 3 deterministic analysis uncertainties as model uncertainty, even those that relate to input parameters in the MELCOR and MACCS consequence models.

- Completeness Uncertainty: Completeness uncertainty arises from limitations in the scope and completeness of the PRA model. These uncertainties can be addressed by supplementing the PRA with additional analyses to demonstrate their impact is not significant. The PRA model may have additional uncertainties from unknown risk contributors, and defense-in-depth principles typically address them. See for example, the discussion in NUREG/KM-0009, "Historical Review and Observations of Defense-in-Depth" (NRC, 2016d). Section 3.1 of NUREG/KM-0009 notes the role of defense-in-depth in a risk-informed regulatory framework to compensate for uncertainties, in particular unquantified and unquantifiable uncertainties. Similar to the framework laid out in Regulatory Guide 1.174 for risk-informed plant-specific changes to licensing bases, consideration of completeness uncertainty means that a regulatory analysis should not be overly reliant on precise risk quantification alone.

Although PRA cannot account for the unknown and identify all unexpected event scenarios, it can (1) identify some originally unforeseen scenarios, (2) identify where some of the uncertainties exist in plant design and operation, and (3) for some uncertainties, quantify the extent of the uncertainty.

NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," issued March 2017 (NRC, 2017a), contains useful general guidance. NUREG-1855 focuses on sources of uncertainty associated with PRAs used to estimate CDF and LERF, since these are the metrics for current risk-informed regulatory decisions, such as risk-informed changes in the licensing basis. However, the principles and broad guidance are more generally applicable to analyses that encompass additional Level 2 (accident progression and source terms) and Level 3 PRA (offsite consequences) information.

Several reference documents contain useful compendiums of sources of uncertainties in Level 2 and Level 3 PRA analyses. An Electrical Power Research Institute (EPRI) companion document to NUREG-1855 lists sources of Level 2 analysis uncertainties identified at a workshop of practitioners (EPRI, 2012). A joint Commission of European Communities expert elicitation conducted in the 1990s identified sources of Level 3 analysis uncertainties (NRC and Commission of European Communities, 1995). The uncertainties for non-site-specific parameters from this expert elicitation were further mapped on to MACCS code input parameters and documented for use in MACCS analyses in NUREG/CR-7161, "Synthesis of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequence Analyses," issued April 2013 (NRC, 2013c). The NRC's Site Level 3 PRA will have companion uncertainty documents for the Level 2 and Level 3 analyses. SOARCA uncertainty analyses are documented for specific SBO scenarios at three NPPs (NRC, 2016b; NRC, 2019a; NRC, 2020). The SOARCA analyses identified and propagated input parameter uncertainties through the MELCOR and MACCS analyses and showed the effects of MELCOR uncertainties on accident progression and radionuclide release metrics, as well as the combined effects of MELCOR and MACCS uncertainties on offsite consequence metrics.

As noted above, NUREG-1855 and the ASME/ANS PRA standard categorize most uncertainties embodied in the Level 2 and Level 3 portions of the PRA as model uncertainties. For the purposes of consequence analyses supporting regulatory analysis, the outputs from MELCOR and MACCS analyses become inputs to the regulatory and cost-benefit analyses as, for example, individual early and latent cancer fatality risk (for QHO comparisons) and averted population dose and offsite economic cost risks (for quantification of benefits to be compared against implementation costs).

It is practical to treat the relevant PRA outputs as parameter uncertainties for cost-benefit analysis. The regulatory bases documents for CPRR (NRC, 2018b) and filtered vents (NRC, 2012h) contain examples of how to characterize and propagate uncertainties. Table 12 of Enclosure 5 to the filtered vents analysis (NRC, 2012h) shows how the uncertainty was described for all relevant inputs to the offsite risk analysis. The point estimates of the base-case inputs such as CDF and MACCS consequences were specified to be the arithmetic means of their respective distributions, and the distribution type and shape factors (such as the α and β parameters for the beta distribution, or the error factor for the lognormal distribution), were specified as well. The staff used a Monte Carlo process to propagate the uncertainty in each of these inputs, as well as the uncertainty in the onsite cost elements. The results are shown for each proposed modification and are presented as the distributions of averted cost (benefit) elements for (1) public dose risk, (2) offsite economic cost risk, (3) onsite worker dose risk, and (4) onsite cost risk. The CPRR risk analysis similarly assigned uncertainty distributions to the following important inputs: the frequency of extended loss of alternating current power events, the seismic hazard curves, the seismic fragility curves, random equipment failures, operator actions, and consequences. The staff used a Monte Carlo process to propagate these uncertainties and show the resulting distribution of individual latent cancer risk for the different regulatory alternatives under consideration (NRC, 2015a, Figure 4-5), which is reproduced as Figure H-6 as an illustrative example.

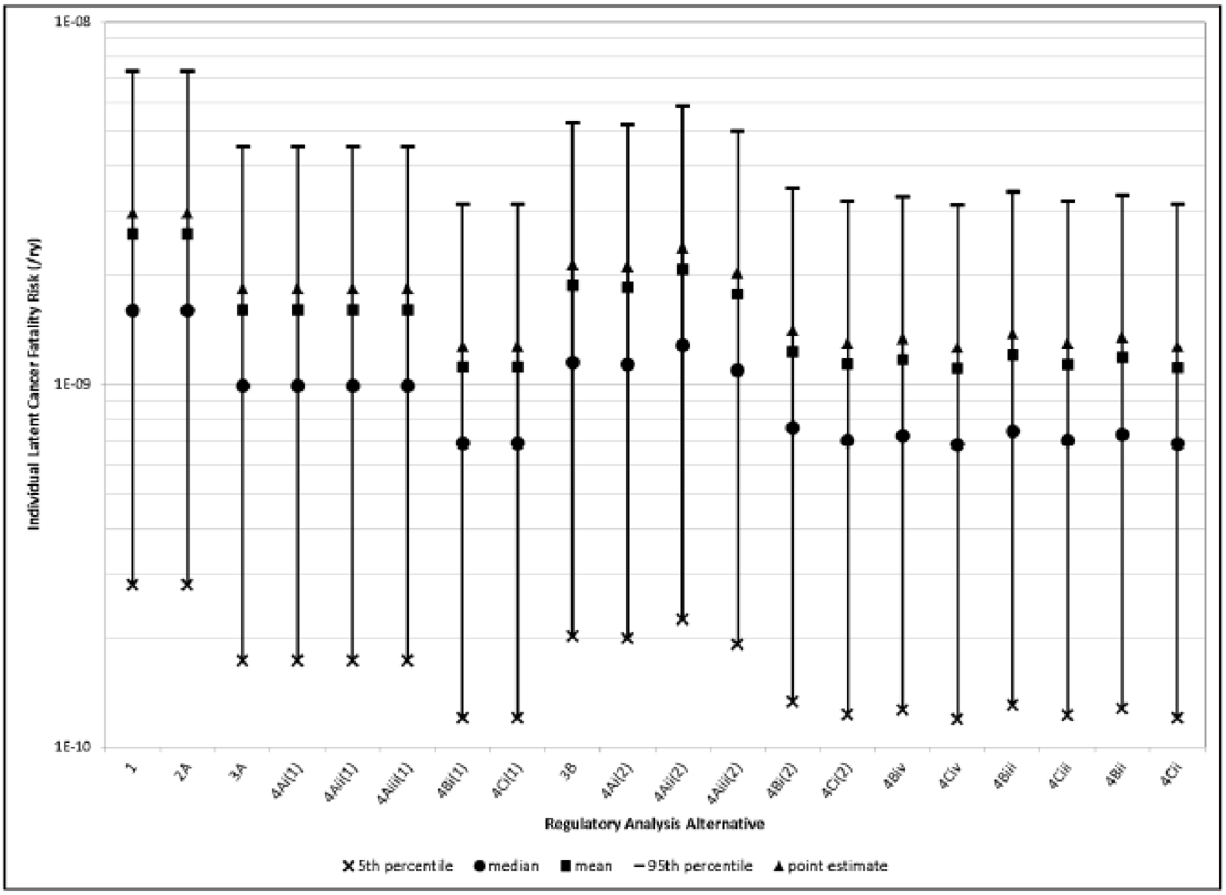


Figure H-6 Parametric Uncertainty Analysis Results for Individual Latent Cancer Fatality Risk

H.6.2 Sensitivity Analyses and Plant-to-Plant Variability Analyses

Sensitivity analysis refers to studying the impact of one uncertain input on the analysis output, without regard to relative probabilities. Uncertainty analysis typically evaluates the integrated impact on the output of a collection of uncertain inputs that are assigned distributions of values, resulting in a distribution of output results. In contrast, sensitivity analysis typically evaluates the impact of one input on the output, and without consideration of the probability of different outcomes. “Two-way” or joint sensitivity analyses similarly can study the impact of two or more uncertain inputs on the outputs of interest.

Sensitivity analyses are typically used for particular categories of inputs. It is more appropriate to use sensitivity, rather than uncertainty, analysis for input values subject to the decisionmaker’s value choices; the dollar per person-rem conversion factor used in cost-benefit analysis is one example. Inputs that depend on variability within the population of affected plants is another example where sensitivity analysis is more appropriate.

H.6.2.1 Sensitivity Analyses

The regulatory analyses discussed in Enclosures H-3 through H-6 of this appendix used sensitivity analyses to address the impact of different values for various inputs. For example, at the time of the filtered vents analysis (Enclosure H-3), CPRR analysis (Enclosure H-4), and expedited spent fuel transfer analysis (Enclosure H-6), the staff was in the process of updating the dollar per person-rem conversion factor. The staff thus performed sensitivity analyses to evaluate the impact on the results of increasing the dollar per person-rem conversion factor from the 1995 value of $2,000 per person-rem to $4,000 per person-rem.

H.6.2.2 Plant-to-Plant Variability Analyses

Variability refers to the inherent heterogeneity of data in an assessment because of the diversity of the regulated facilities. When conducting an analysis for a generic requirement that would apply to a number of different plants, the staff usually chooses a representative plant and site for the base-case analysis. To assess the potential difference in analysis outcomes for the affected variable population of sites and facilities, the staff should complete a plant-to-plant variability analysis. For example, the expedited spent fuel transfer regulatory analysis (NRC, 2013g) and technical basis (NRC, 2014d), as well as the CPRR analysis (NRC, 2015a; NRC, 2018b), included sensitivity analyses that showed the effect of the same accident occurring at different sites.

For the CPRR analysis, the staff performed MACCS sensitivity calculations to analyze the influence of site-to-site variations and protective action variations on the offsite consequences. The staff conducted the following sensitivity calculations:

- population (low, medium, high)
- evacuation delay (1 hour, 3 hour, 6 hour, no evacuation)
- nonevacuating cohort size (5 percent of EPZ population)
- intermediate phase duration (0, 3 months, and 1 year)
- long-term habitability criterion (500 millirem per year and 2 rem per year), which can vary among states in the United States

Table H-5 shows one example of results from this set of sensitivity calculations. This table shows the ratio of results if the intermediate phase duration were 1 year instead of the baseline duration of 3 months. The color coding visually shows the significance to various metrics. Yellow indicates a ratio of near 1, meaning there was no significant difference, while colors closer to red or green indicate a larger influence on results. Results are reported for three sites with representative low, medium, and high populations, coupled with low, medium, and high source terms for Mark I and Mark II containments. Table H-5 shows that the conditional offsite costs for the high source terms at all six sites evaluated are approximately 1.6 times higher when the intermediate phase is assumed to last for 1 year versus 3 months.

Table H-5 Ratio of Consequences for 1-Year Intermediate Phase Duration Sensitivity Cases to Baseline Cases in the Containment Protection and Release Reduction Analysis

Base Model	Site	Source Term	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Low - Hatch	Mark I - Low (Bin 3)	Individual early fatality risk is zero for all baseline and sensitivity cases.	0.88	0.89	0.88	0.98	0.99	0.98	0.98	1.00	1.00	0.00	0.00
		Mark I - Med (Bin 10)		1.07	0.93	0.91	0.97	0.97	1.38	1.18	0.86	0.92	0.48	0.48
		Mark I - High (Bin 17)		1.04	0.98	0.93	0.98	0.96	1.61	1.39	0.80	0.87	0.60	0.53
	Medium - Vermont Yankee	Mark I - Low (Bin 3)		0.88	0.88	0.88	0.94	0.95	0.96	0.96	1.00	1.00	0.14	0.14
		Mark I - Med (Bin 10)		1.06	0.92	0.89	0.93	0.92	1.39	1.04	0.73	0.86	0.57	0.57
		Mark I - High (Bin 17)		1.02	0.97	0.91	0.97	0.92	1.58	1.33	0.71	0.82	0.59	0.46
	High - Peach Bottom	Mark I - Low (Bin 3)		0.88	0.89	0.88	0.95	0.95	0.97	0.97	1.00	1.00	0.16	0.16
		Mark I - Med (Bin 10)		1.07	0.92	0.90	0.93	0.92	1.31	1.16	0.91	0.94	0.39	0.39
		Mark I - High (Bin 17)		1.04	0.97	0.93	0.97	0.94	1.60	1.46	0.86	0.89	0.55	0.51
Mark II - Limerick	Low - Columbia	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.99	0.99	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		0.96	0.92	0.92	0.98	0.98	1.00	1.00	0.99	1.00	0.29	0.29
		Mark II - High (Bin 8)		1.18	0.98	0.98	0.98	0.98	1.50	1.49	0.86	0.90	0.20	0.19
	Medium - Susquehanna	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.96	0.96	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		0.98	0.93	0.90	0.95	0.93	1.18	1.11	0.94	0.97	0.44	0.44
		Mark II - High (Bin 8)		1.18	0.98	0.98	0.97	0.97	1.63	1.49	0.62	0.81	0.26	0.21
	High - Limerick	Mark II - Low (Bin 2)		0.90	0.93	0.93	0.93	0.94	1.00	1.00	1.00	1.00	*	*
		Mark II - Med (Bin 5)		1.00	0.92	0.91	0.94	0.93	1.08	1.06	0.96	0.97	0.45	0.45
		Mark II - High (Bin 8)		1.17	0.97	0.98	0.95	0.96	1.57	1.48	0.68	0.81	0.21	0.20

* An asterisk indicates that the values of both the numerator and denominator in the ratio are zero.
(Source: NUREG-2206, Table 4-33)

H.7 PRESENTATION OF RESULTS—INPUTS TO REGULATORY ANALYSIS

H.7.1 Aggregating Probabilistic Risk Assessment Results from Different Hazards

For many regulatory applications, it is necessary to consider the contributions from several hazards to a specific risk metric. When considering multiple hazards, a PRA model can be a fully integrated model in which all hazards are combined into a single logic structure, a set of individual PRA models for each hazard, or a mixture of the two. When combining the results of PRA models for several hazards, the levels of detail and approximation included in the PRA model may differ from one hazard to the next. Because of the methods and data used, a high level of uncertainty can exist in PRAs for internal fires, external events (seismic, high wind, and others), and low-power/shutdown conditions. In principle, this uncertainty could be reduced by developing models to the same level of detail and rigor associated with internal events, at-power PRAs. A larger uncertainty in a subset of the total PRA analyses can result in greater uncertainty. The analyst needs to understand the main sources of conservatism in the PRA associated with any of the hazards that can potentially impact the regulatory application. When interpreting the results of the comparison of risk metrics to acceptance criteria or guidelines, it is important to focus not only on the aggregated numerical result but also on the relative importance and uncertainty of the main contributors to the risk metric.

H.7.2 Offsite Consequence Measures

An analyst uses several offsite consequence measures to characterize the impacts resulting from a severe accident. For the purposes of a regulatory analysis, the individual early fatality risk, latent cancer fatality risk, population dose, and offsite economic costs should all be presented. The first two enable comparisons with the NRC's QHOs, and the latter two are needed to quantify the affected parameters (accident offsite consequences) in the cost-benefit equation.

H.7.2.1 Conditional Consequence Measures

Conditional offsite consequence results should be presented, first, for each source term bin. In other words, given that an accident occurs and results in a particular source term bin, the offsite consequences should be presented. The next step is to map the source term bins onto the release categories developed in the accident sequence analysis, for the purposes of risk integration.

Early Fatality Risk

Individual early fatality risk for the area within approximately 1 mile of the site boundary is provided as an input for the evaluation of the NRC's early fatality QHO (NRC, 2015a).[22]

Latent Cancer Fatality Risk

The individual latent cancer fatality risk is the risk of an average individual within the specified spatial element contracting a fatal cancer caused by early, intermediate, and long-term radiation

[22] If no one resides within 1 mile of the site boundary an individual should be assumed to reside within 1 mile for evaluation purposes.

exposures. The analyst calculates this population-weighted metric by dividing the expected number of fatal cancers in a spatial element by the population residing in that element. The analysis should show the individual latent cancer fatality risk for the areas within 10- and 50-miles from the reactor site. The 10-mile area corresponds to the QHO for cancer fatality risk (NRC, 2015a) and to the plume exposure EPZ. The analysis also should display the results for the 50-mile area, as the NRC's regulatory analyses use this distance (other distances may be appropriate, depending on facility type, as discussed in Section H.3.3.3).

Population Dose Risk

The offsite population dose, measured in person-rem, represents the sum of the doses from all exposure pathways multiplied by the size of the population within a specified area. This metric quantifies the public health (accident) attribute, as discussed in Sections 5.2.1 and 5.3.2.1 of this NUREG. The dose to the population within a 50-mile radius (or other appropriate distance, as discussed in Section H.3.3.3) from the reactor facility is reported for each source term bin. MACCS reports the population dose per event (i.e., the conditional dose, given a particular accident), and this value needs to be converted to the population dose per reactor-year by multiplying by the event frequency.

Offsite Economic Cost Risk

The offsite economic costs resulting from an accident scenario correspond to the economic consequences (offsite property) attribute described in Sections 5.2.5 and 5.3.2.5 of this NUREG. This metric sums the costs of the protective actions taken to reduce offsite exposure and restore land to usability and habitability. The offsite economic costs are computed directly by MACCS and should be reported for the area within a 50-mile radius (or other appropriate distance, as discussed in Section H.3.3.3) of the reactor facility for each source term bin.

Other Results

In addition to risk estimates, other consequence results provide risk insights about the various alternatives. Some examples include the number of displaced individuals, land contamination, and the extent over which protective actions may be needed. Discussion of these other results may provide a better understanding of the extent and severity of the accident scenarios.

Table H-6 gives one example of how this information might be tabulated. This table is taken from the CPRR analysis (NRC, 2015a; NRC, 2018b) and shows each of these consequence results and their corresponding source term bins. This CPRR analysis (similar to the SFP study [NRC, 2014d]) reported other results, such as land contamination and size of the population affected by long-term protective actions, at radii of 50 miles and 100 miles from the reactor site.

Table H-6 Severe Accident Consequence Analysis Results—Example

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	0	9.72E-08	1.03E-08	3.45E-09	282	345
2	5DF1000	0.0006%	0.005%	32.2	20	0	1.15E-06	1.81E-07	6.35E-08	4,340	5,440
3	42DF100	0.0043%	0.037%	14.3	13	0	6.58E-06	8.67E-07	3.02E-07	20,700	26,700
4	11	0.042%	0.45%	20.3	20	0	7.90E-05	9.68E-06	3.27E-06	202,000	261,000
5	51DF10	0.23%	2.01%	16.6	9	0	1.35E-04	3.39E-05	1.21E-05	689,000	888,000
6	5	0.55%	4.94%	32.2	20	0	2.29E-04	1.05E-04	4.01E-05	2,160,000	2,900,000
7	3	1.09%	10.26%	14.3	20	0	3.08E-04	1.88E-04	7.43E-05	4,140,000	5,580,000
8	1	2.46%	19.81%	22.8	25	0	4.70E-04	3.17E-04	1.25E-04	6,110,000	8,260,000
9	52	3.57%	28.67%	16.6	10	0	4.03E-04	2.46E-04	1.01E-04	5,430,000	7,440,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	381,000,000	381,000,000	-	-	-	-
2	5DF1000	0.0006%	0.005%	32.2	20	381,000,000	381,000,000	0	0	-	-
3	42DF100	0.0043%	0.037%	14.3	13	393,000,000	393,000,000	2	2	0	0
4	11	0.042%	0.45%	20.3	20	844,000,000	846,000,000	44	47	1,030	1,030
5	51DF10	0.23%	2.01%	16.6	9	4,250,000,000	4,380,000,000	130	221	15,400	15,400
6	5	0.55%	4.94%	32.2	20	24,000,000,000	28,000,000,000	303	551	62,400	62,400
7	3	1.09%	10.26%	14.3	20	80,800,000,000	105,400,000,000	698	1,200	619,000	649,000
8	1	2.46%	19.81%	22.8	25	85,500,000,000	109,300,000,000	854	1,680	721,000	741,000
9	52	3.57%	28.67%	16.6	10	53,600,000,000	63,800,000,000	618	1,400	414,000	449,000

(Source: SECY-15-0085, Enclosure, Table 4-22)

The consequence results presented in Table H-6 do not account for the event frequency, (e.g., they are conditional on the occurrence of the postulated accident). Also, it is important to note that these results are strongly dependent on the assumed (modeled) protective actions.

H.7.3 Evaluation of Regulatory Alternatives

H.7.3.1 Results from the Core Damage Event Tree Quantification

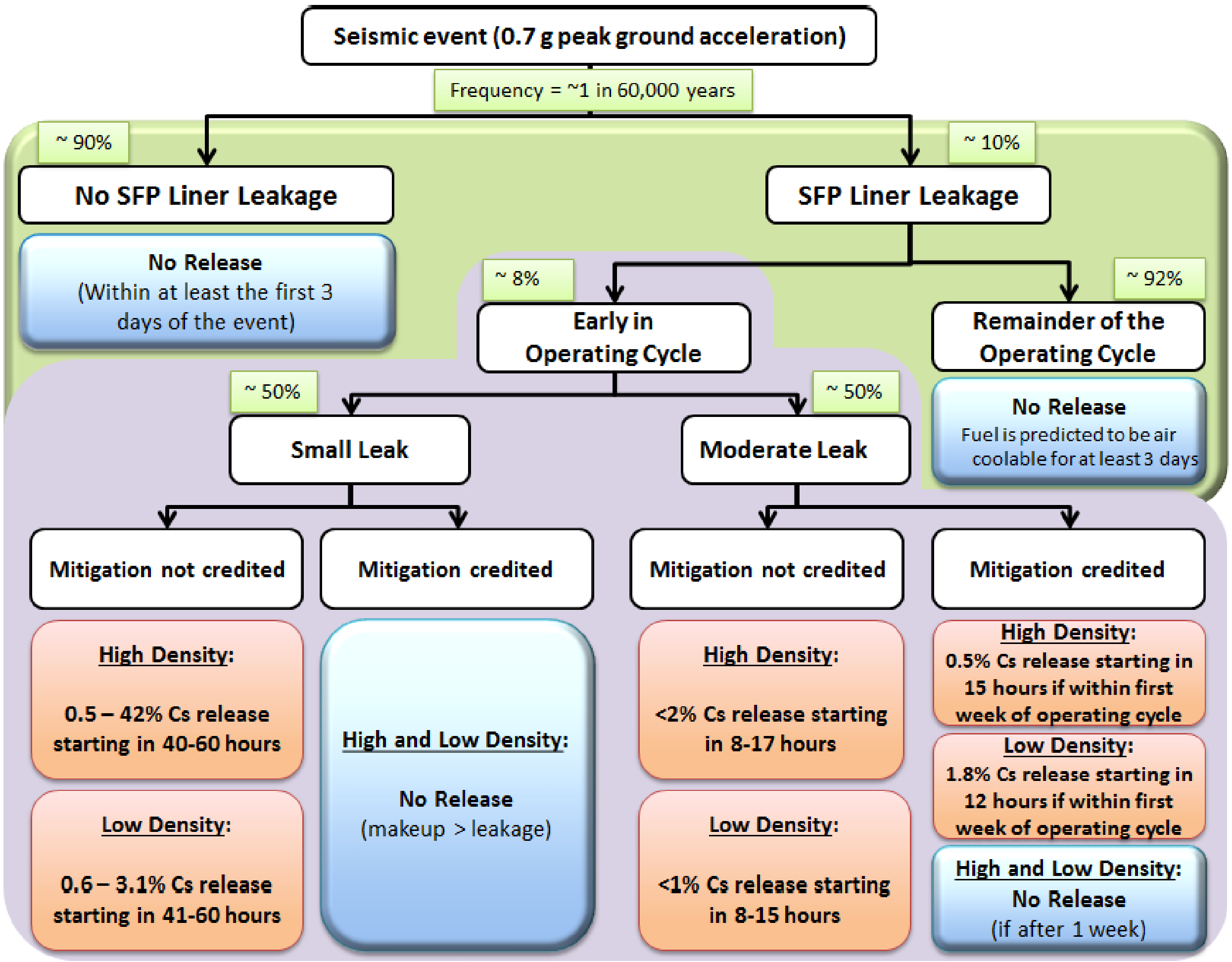
The analysis should tabulate the point estimates for relevant initiating event frequency, CDF, and conditional core damage probability by site for each regulatory alternative. These tables provide insight into the efficacy of the different strategies and present fleet averages for CDF and conditional core damage probability for comparison.

Basic events, such as equipment and human failure events, should be tabulated with importance measures (Risk Achievement Worth and Fussel-Vesely) with respect to CDF. A table should show plant damage state frequencies for each regulatory alternative.

H.7.3.2 Results from the Accident Progression Event Tree Quantification

The analysis should tabulate the conditional containment failure probability for each APET to demonstrate the efficacy of different mitigation alternatives. It should also tabulate the frequencies of significant release categories for each APET.

The accident sequence analysis results show the CDF frequency from the initiating event and provide insights into the relative contributions of various factors (e.g., external hazards, equipment failures, human errors) to overall CDF. Figure H-7 shows an example of accident sequence analysis and radioactive release summary results from the SFP study (NRC, 2014d).



Note: The low-density pool has about 1/3 of Cs-137 inventory compared to high-density pool. Early in the operating cycle refers to early time after shutdown.

Figure H-7 Likelihood of a Leak and Magnitude of Releases from Beyond-Design-Basis Earthquake

(Source: NUREG-2161, Figure ES-1)

H.7.3.3 Results from MELCOR Analysis

The MELCOR results are classified into two broad categories: (1) thermal-hydraulic output and (2) source term output. The timing of key events for the accident progression should be presented and discussed for select MELCOR cases. In addition, time plots should be provided for some important thermal-hydraulic outputs. Some examples include the following:

- Reactor pressure vessel pressure, temperature, and water level

- Containment pressure and temperature, to determine the likelihood of failure of containment and various components by overpressure, overtemperature, or both

- Hydrogen and other noncondensable gas generation and migration, to contribute to containment overpressurization; also, to determine the potential for combustion in, for example, the reactor building or the vent line

These discussions assist the analyst in assessing how each regulatory alternative would impact the accident progression and the state of containment vulnerability under severe accident conditions. They also provide the decisionmaker with qualitative information and a technical basis for developing potential staff guidance for implementing a regulatory alternative.

H.7.4 Risk Integration Results and Key Insights

The final step is to present the results as integrated risk measures, which multiplies the frequencies of different accident sequences with their conditional consequences. For example, for each regulatory alternative (or subalternative), the population dose risk and offsite economic cost risks should be presented on a per-reactor-year basis. Table H-7 and Figure H-8 show example presentations of results, taken from the CPRR analysis (NRC, 2015a; NRC, 2018b). The affected parameters that are quantified in the cost-benefit equation, population dose risk, and economic cost risk, associated with each regulatory analysis subalternative are presented for 50-mile and 100-mile radial distances. Additional measures are also presented, such as land exceeding habitability criterion. Figures H-9 and H-10 show another example, taken from the filtered vents analysis (NRC, 2012h), which presents the change (compared to the status quo) in offsite economic cost risk per year for each regulatory alternative, called a Mod (Figure H-9). Furthermore, the results of the uncertainty quantification are shown for those alternatives (Figure H-10) with a positive change.

In addition to quantitative risk results, important qualitative insights and assumptions should also be presented, on the most important contributors to risk and uncertainty. The supplementary analyses discussed in Section H.6 make an essential contribution to this summary discussion for decision makers, since those investigations help identify the impact of uncertainties and the sensitivity of results to different assumptions. For example, the Technical Evaluation Summary of the CPRR analysis (NRC, 2015a, Section 4.6 of Enclosure) presented the key insights from the risk evaluation, MELCOR analysis, and MACCS analysis. These insights included the following:

- A discussion of the most important contributors to accident frequency (e.g., the major contribution to seismically induced ELAP is from earthquakes that cause site peak ground accelerations in the range of 0.3 to 0.75g)

- A discussion of important assumptions (e.g., the evaluation assumed that 60 percent of the time, the pre-core-damage water addition [FLEX] will be successful in preventing core damage)

- A discussion of accident progression and source term insights (e.g., the highest calculated release to the environment results from a main steam line creep rupture scenario, which is one of the least likely scenarios)

- A discussion of offsite consequence insights (e.g., that, for all Mark I and Mark II source terms, there is zero early fatality risk because the source terms are not large enough to exceed the threshold for the acute dose to the red bone marrow, which is typically the most sensitive tissue for early fatalities)

- A discussion of important uncertainties and their key drivers (e.g., that the 5 percent/95 percent parametric uncertainty interval of the estimated risks is more than 1 order of magnitude and is largely driven by uncertainty in the seismic hazard curves)

Table H-7 Risk Estimates by Regulatory Analysis Subalternative

Index	Regulatory Analysis Sub-Alternative	Fraction of Core-Damage Frequency		Individual Early Fatality Risk (/y)	Individual Latent Cancer Fatality Risk (/y)			Population Dose (person-rem/y)		Offsite Cost ($ 2013/y)		Land Exceeding Long-Term Habitability Criterion (square miles/y)		Population Subject to Long-Term Protective Actions (persons/y)	
		Vented	Uncontrolled Release	0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	1	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
2	2A	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
3	3A	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
4	3B	42%	58%	0.0E+00	2.1E-09	6.7E-10	3.4E-10	1.1E+01	1.9E+01	7.4E+04	1.0E+05	3.4E-03	6.4E-03	4.1E-01	4.9E-01
5	4Ai(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
6	4Ai(2)	42%	58%	0.0E+00	2.1E-09	6.1E-10	3.1E-10	9.5E+00	1.7E+01	6.8E+04	9.0E+04	3.2E-03	5.8E-03	3.6E-01	4.1E-01
7	4Aii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
8	4Aii(2)	42%	58%	0.0E+00	2.4E-09	7.7E-10	3.9E-10	1.2E+01	2.2E+01	8.9E+04	1.2E+05	3.9E-03	7.3E-03	4.8E-01	5.8E-01
9	4Aiii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
10	4Aiii(2)	42%	58%	0.0E+00	2.0E-09	5.6E-10	2.7E-10	8.7E+00	1.5E+01	6.2E+04	7.9E+04	3.0E-03	5.1E-03	3.1E-01	3.4E-01
11	4Bi(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
12	4Bi(2)	42%	58%	0.0E+00	1.4E-09	3.3E-10	1.5E-10	4.8E+00	8.2E+00	3.1E+04	3.8E+04	1.8E-03	2.7E-03	1.6E-01	1.6E-01
13	4Bii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.6E+00	7.9E+00	3.0E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
14	4Biii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.7E+00	8.1E+00	3.1E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.6E-01
15	4Biv	40%	60%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.6E+00	7.8E+00	3.0E+04	3.6E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
16	4Ci(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
17	4Ci(2)	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.5E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
18	4Cii	42%	58%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.4E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01
19	4Ciii	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.4E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
20	4Civ	40%	60%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.3E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01

(Source: NUREG-2206, Table 5-1)

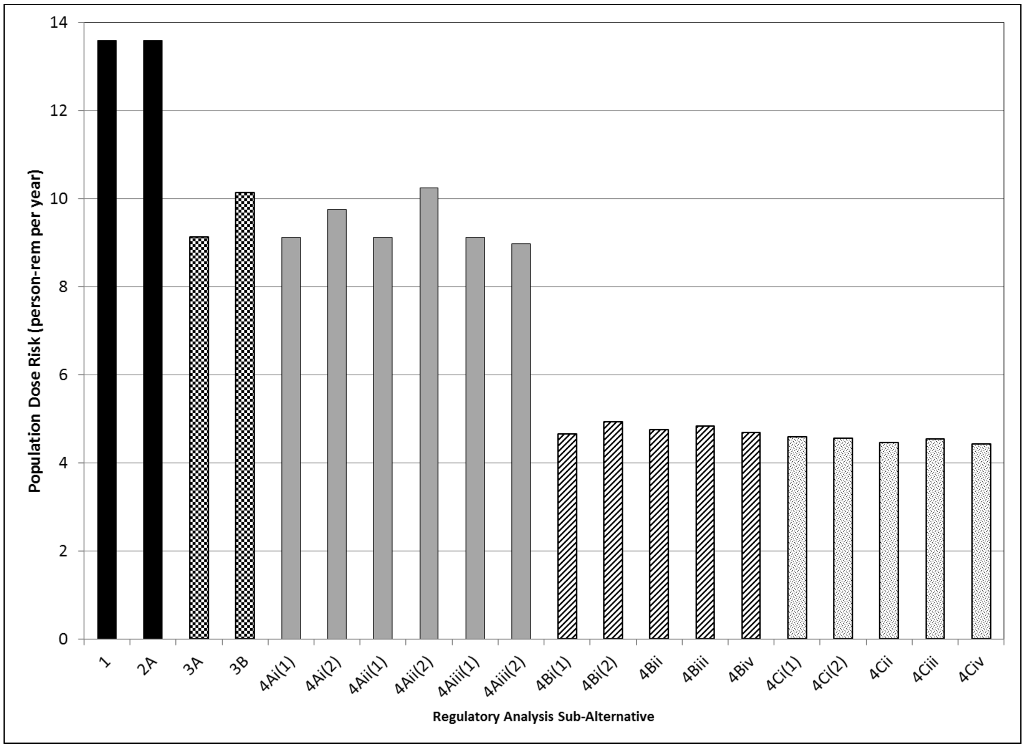


Figure H-8 Comparison of Regulatory Analysis Alternatives Using Population Dose Risk (0-50 miles)

(Source: NUREG-2206, Figure 5-2)

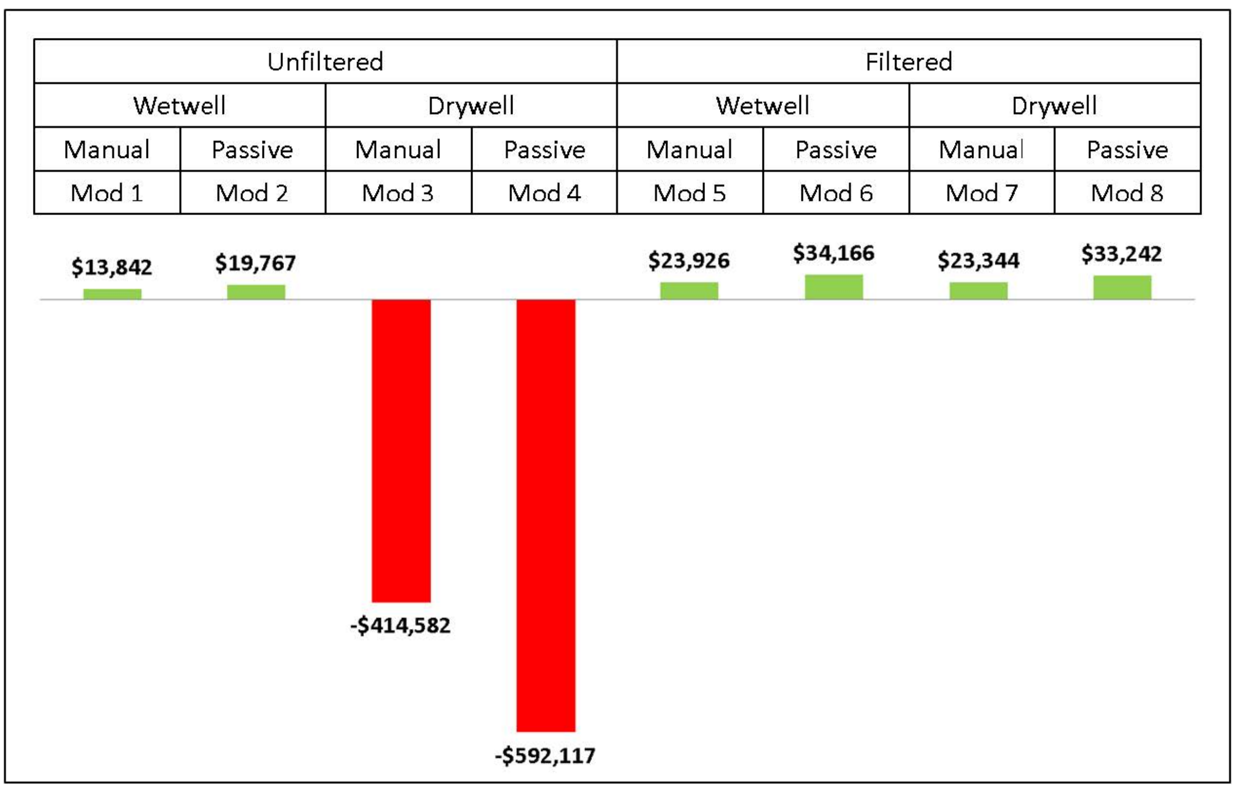


Figure H-9 Reduction in 50-mile Offsite Cost Risk (Δ$/reactor-year)

(Source: SECY-12-0157, Enclosure 5c, Figure 5)

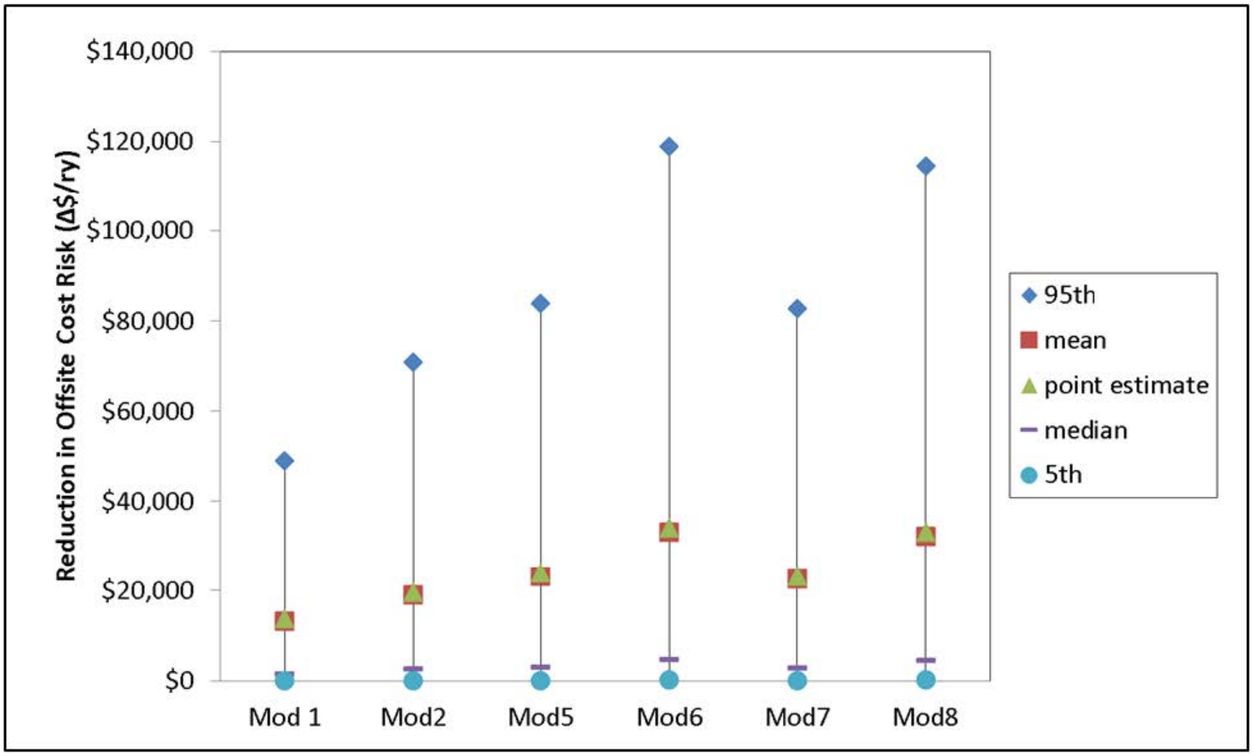


Figure H-10 Uncertainty in Reduction in 50-mile Offsite Cost Risk
(Source: SECY-12-0157, Enclosure 5c, Figure 10)

H.8 REFERENCES

42 USC 2011 et seq. Atomic Energy Act of 1954, as amended.

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ENCLOSURE H-1: DESCRIPTION OF ANALYTICAL TOOLS AND CAPABILITIES

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, probabilistic risk assessments (PRAs) for nuclear power plants (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) radiological release, and (3) offsite radiological consequences.

Several computer codes exist for performing PRA and severe accident consequence analysis. For regulatory analyses that require detailed analyses of offsite consequences, most recent light-water reactor applications have used the U.S. Nuclear Regulatory Commission (NRC)-sponsored MELCOR and MELCOR Accident Consequence Code System (MACCS) code suites. These codes include state-of-the-art integrated modeling of severe accident behavior that incorporates insights from decades of research into severe accident phenomenology and radiation health effects. The NRC-sponsored Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE) code is also available for performing PRAs using event trees and fault trees. Figure H-11 notes the role of these three code suites in NPP PRAs. The sections below describe these code suites, their capabilities, and their typical uses.

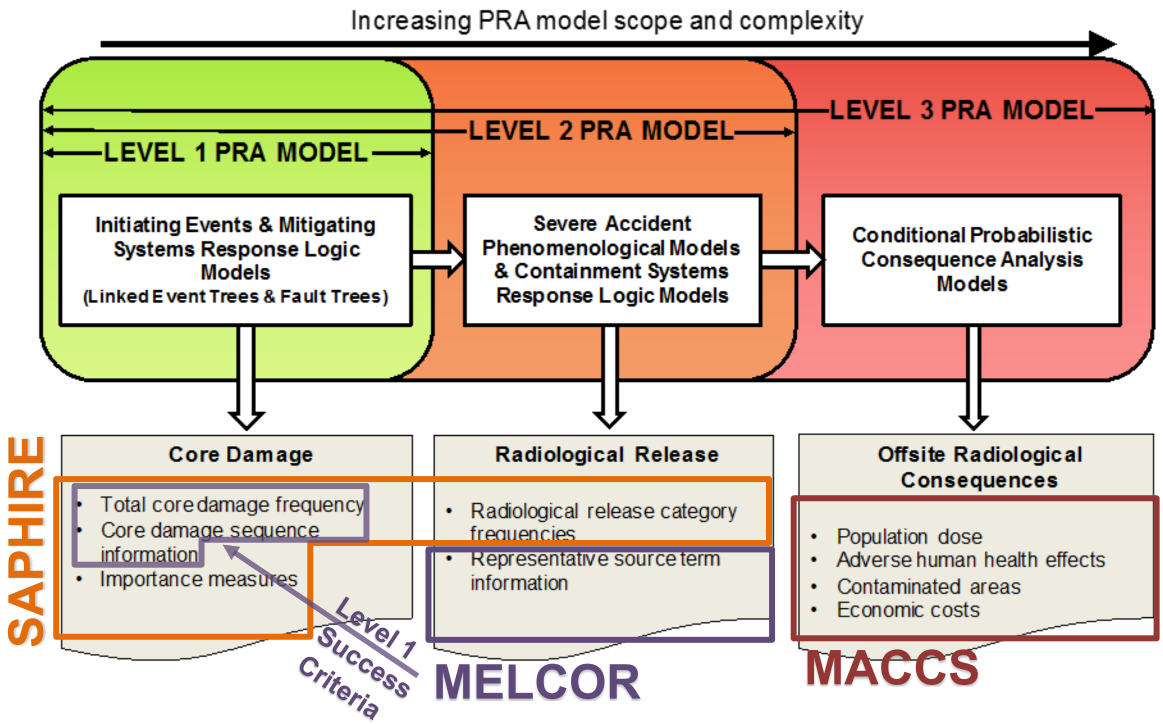

Figure H-11 Overall Logic and Structure of Traditional NPP PRA Models and Role of SAPHIRE, MELCOR, and MACCS Code Suites

Severe Accident Scenario Modeling and Frequency Analysis

Systems Analysis Programs for Hands-on Integrated Reliability Evaluations (SAPHIRE)

SAPHIRE is an NRC-sponsored software application that the Idaho National Laboratory developed and maintains for performing PRAs of complex engineered facilities, systems, or processes.

The NRC uses SAPHIRE to develop Level 1 and Level 2 PRA logic models for NPPs. The end state of interest for a Level 1 PRA is core damage. SAPHIRE can (1) model plant and operator responses to initiating events to identify sequences (combinations of system and operator action successes and failures) that result in either the achievement of a safe state or the onset of core damage, (2) quantify the frequencies of sequences that result in core damage and total core damage frequency (CDF) for the NPP, and (3) identify important contributors to CDF. The end state of interest for a Level 2 PRA is radiological release. SAPHIRE can also be used to expand upon a Level 1 PRA model to (1) model containment systems and operator responses to severe accident conditions, (2) quantify radiological release category frequencies—including a large early release frequency (LERF), and (3) identify important contributors to radiological release category frequencies. A Level 3 PRA combines the results of the SAPHIRE radiological release category frequencies (from the Level 2 PRA) with the results from the corresponding MACCS offsite radiological consequence model to provide an overall characterization of the risk to the offsite public from a broad spectrum of postulated accidents involving a modeled NPP site.

SAPHIRE contains graphical editors for creating, viewing, and modifying fault tree and event tree models that serve as logical representations of accident sequences that can occur at an NPP. SAPHIRE uses event tree and fault tree models, coupled with accident sequence linkage rules and postprocessing rules, to generate unique combinations of individual failures (i.e., minimal cut sets) that can result in an undesired end state. SAPHIRE quantifies the frequencies and probabilities associated with the minimal cut sets to estimate the frequencies of selected undesired end states. In addition, SAPHIRE includes many useful features to support the frequency quantification of PRA models and identification of significant contributors to risk (e.g., calculation of traditional PRA importance measures described below). Finally, SAPHIRE can perform an uncertainty analysis using either Monte Carlo or Latin Hypercube sampling methods to estimate the uncertainty in calculated results (e.g., CDF, LERF, or importance measures) caused by epistemic[23] uncertainties in input parameters for basic events in the Level 1 and Level 2 PRA logic models.

NUREG/CR-7039, "Systems Analysis Programs for Hands-on Integrated Reliability Evaluations Version 8," issued June 2011, contains detailed information about the features and capabilities of SAPHIRE Version 8. Some basic features and capabilities in SAPHIRE include the following:

- Basic events: Basic events typically represent events involving failures of structures, systems, or components; adverse environmental or phenomenological conditions that could lead to failures; or human failure events for operator actions. Basic events are logically linked together in fault trees and provide SAPHIRE with the probabilistic information (e.g., failure data input and type of probability calculation) needed to quantify the PRA model. Basic events appear as circles at the bottom of the example in Figure H-12 (feeding System A and System B fault trees).

- Fault trees: A fault tree generally represents a failure model. A fault tree model consists of a top event (e.g., failure of System A in the example in Figure H-12), usually defined by a heading in an event tree (e.g., System A appears as a heading in the example event tree in Figure H-12, for the initiating event "IE"). A combination of basic events must occur to result in the undesired top event, using a logic structure as a model for the basic events.

- Event trees: An event tree is a logic structure that chains sequential events together to model the likelihood of the potential outcome(s) of those events. The simple example in Figure H-12 contains a chain of three events: initiating event "IE," System A (success or failure), and System B (success or failure). The analyst defines accident sequences using an event tree to indicate the failure or success of top events. Each heading in the event tree is associated with a system fault tree. Event trees are constructed and modified using a graphical editor that allows the linkage of multiple event trees and the creation of very large event trees.

- Rule-based fault tree linking: In generating accident sequences, the analyst uses a set of defined rules to reduce the complexity of the overall logic structure.

- Cut sets: A cut set is a combination of faults that must occur together to result in the failure of a top event. To solve an accident sequence, SAPHIRE constructs a fault tree

[23] Epistemic uncertainty is the uncertainty related to the lack of knowledge or confidence about the system or model and is also known as state-of-knowledge uncertainty (NUREG-2122, "Glossary of Risk-Related Terms in Support of Risk-Informed Decision Making," issued November 2013).

for those systems that are defined to be failed in the sequence logic by creating a temporary “AND” gate with these systems as inputs. SAPHIRE then solves this fault tree using specified cut set probability truncation values. This process results in a list of cut sets for the failed systems in the accident sequence. SAPHIRE then uses Boolean reduction techniques to further reduce this list of cut sets to the set of minimal cut sets for the accident sequence. The analyst can specify one of three main cut set quantification techniques, depending on the desired tradeoff between accuracy and computation time.

- Uncertainty analysis: Both Monte Carlo and Latin Hypercube sampling methods are available for performing an uncertainty analysis. The uncertainty analysis functions in SAPHIRE estimate the uncertainty in calculated output quantities caused by epistemic uncertainties in the basic event frequencies or probabilities. These output quantities include (1) fault tree top event probabilities, (2) event tree sequence frequencies, (3) end state frequencies, or (4) importance measures. In an uncertainty analysis, SAPHIRE samples analyst-specified distributions for each basic event in a group of cut sets and then quantifies these cut sets using the sampled values.

- Importance measures: SAPHIRE can quantify a range of traditional importance measures that are used to measure the absolute or relative importance of basic events in the PRA model to specified end-state frequencies. As previously stated, uncertainty analyses on these measures can use Monte Carlo or Latin Hypercube sampling techniques.

The NRC designed its SAPHIRE software development and maintenance program to provide an analytical tool that performs risk calculations accurately and efficiently and reports the results in a clear and concise manner to support risk-informed decisionmaking. Idaho National Laboratory has created a software quality assurance program to ensure SAPHIRE continues to meet its requirements as new features and changes are implemented.

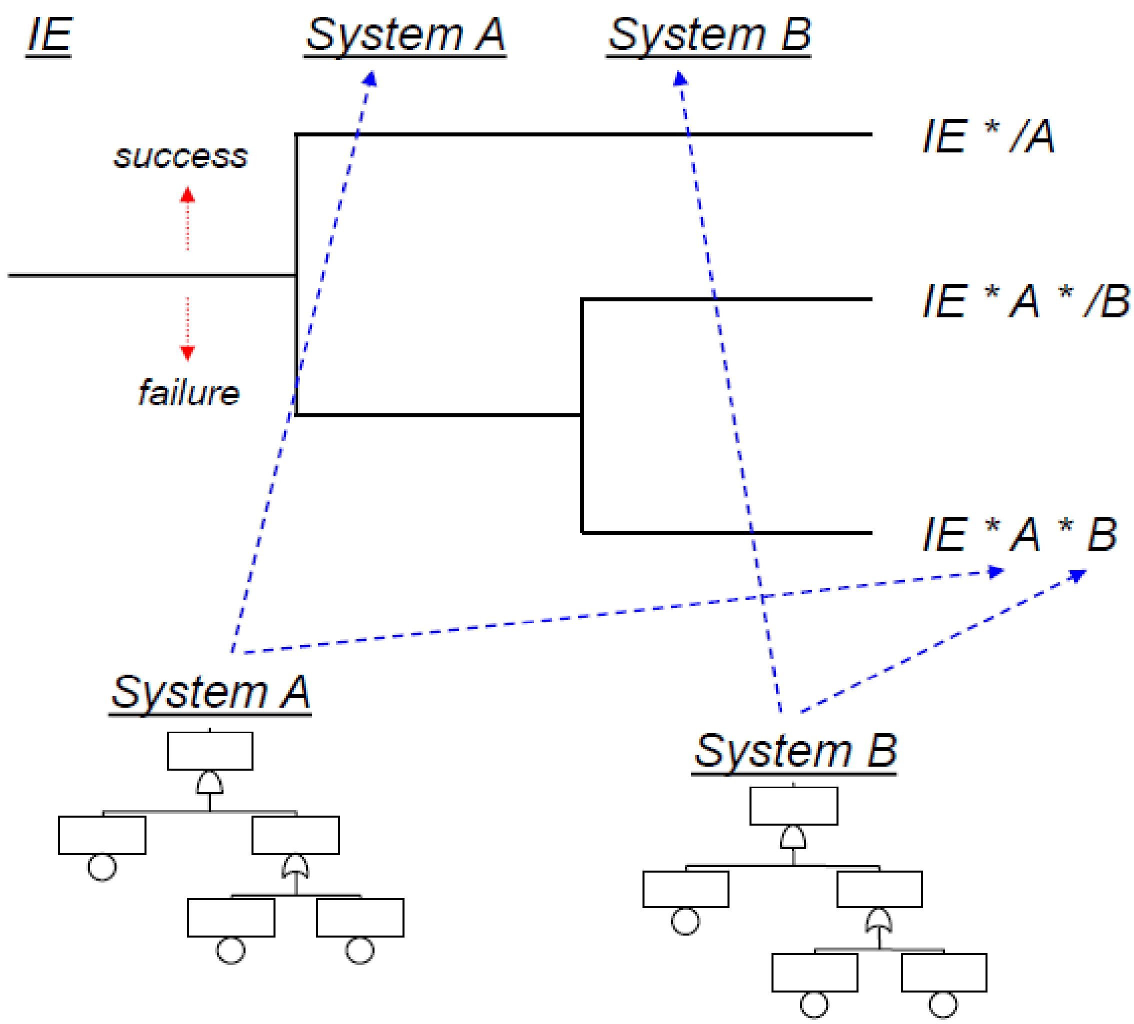

Figure H-12 Simplified Diagram of Event Tree with Initiating Event (IE) and Two Supporting Fault Trees

Standardized Plant Analysis Risk Models

The NRC established the Standardized Plant Analysis Risk (SPAR) model program to support regulatory reviews and independent evaluations of risk-related issues. The SPAR models are plant-specific NRC-developed PRA models using standardized modeling conventions and data. This standardization allows agency risk analysts to efficiently use SPAR models for diverse plant designs in support of various regulatory activities. The regulatory uses of SPAR models include the following:

- Inspection Program (e.g., Significance Determination Process Phase 3): Determine the risk significance (with respect to CDF and LERF) of inspection findings or of events to decide (1) the allocation and characterization of inspection resources, (2) the initiation of an inspection team, or (3) the need for further analysis or action by other agency organizations.

- Management Directive 8.3, “NRC Incident Investigation Program”: Estimate the risk significance of events or conditions at operating NPPs so the agency can analyze and evaluate the implications of plant operating experience to (1) compare the operating experience with the results of licensee PRAs, (2) identify risk-significant conditions that need additional regulatory attention, (3) identify conditions that need less regulatory attention, and (4) evaluate the risk impact of regulatory or licensee programs.

- Accident Sequence Precursor Program: Screen and analyze operating experience data using a systematic approach to identify those events or conditions that are precursors to severe accident sequences (core damage events).

- Generic Issues Program: Provide the capability to resolve generic safety issues, both for screening (or prioritization) and conducting a more rigorous analysis to (1) determine if licensees should be required to make a change to their plants or (2) assess if the agency should modify or eliminate one or more existing regulatory requirements.

- License Amendment Reviews: Enable the NRC staff to make risk-informed decisions on plant-specific changes to the licensing basis as proposed by licensees and provide risk perspectives in support of agency reviews of licensee submittals.

- Verification of Performance Indicators: Assist in (1) identifying threshold values for risk-based performance indicators and (2) developing integrated or aggregate performance indicators.

- Special Studies: Undertake various studies in support of risk-informed regulatory decisions (e.g., regulatory analysis and backfit analysis).

- Operating Experience: Support and provide rigorous and peer reviewed evaluations of operating experience, thereby demonstrating the agency’s ability to analyze operating experience independently of licensee PRAs and thus enhancing the technical credibility of the agency.

The SPAR models allow agency risk analysts to perform independent evaluations of regulatory issues without reliance on licensee-developed PRA models and analyses. The SPAR models integrate systems analysis, accident scenarios, component failure likelihoods, and human reliability analysis into a coherent model that reflects the design and operation of a specific plant. These models give agency risk analysts the capability to (1) quantify the expected risk of an NPP in terms of CDF or LERF, (2) identify and understand the attributes that significantly contribute to risk, and (3) develop insights on how to manage that risk.

The SPAR models use an NRC-developed standard set of event trees and standardized input data for initiating event frequencies, equipment performance, and human performance. However, these input data may be modified to be more plant- or event-specific, when needed. The system fault trees contained in the SPAR models are generally not as detailed as those contained in licensee PRA models. However, SPAR models may need to be more advanced in some areas than licensee PRA models (e.g., modeling of support system initiating events and electrical power recovery). The staff has performed detailed cut set reviews for all SPAR models to (1) more accurately model plant operation and configuration and (2) identify significant differences between licensee PRAs and the corresponding SPAR models.

In addition to internal events, at-power models, the staff has developed the following models for a subset of units: (1) external event models based on the licensee responses to Generic Letter 88-20, Supplement 4, “Individual Plant Examination of External Events for Severe Accident Vulnerabilities,” dated June 28, 1991, (2) low-power/shutdown models, and (3) extended Level 1 PRA models supporting limited Level 2 PRA modeling and quantification of LERF. SPAR model development work in these areas is ongoing. The staff has updated all internal events models to include FLEX modeling. Additionally, the staff has developed design-specific internal events SPAR models for new reactor designs and is developing a plant specific new reactor SPAR model.

The staff has developed a formal SPAR model quality assurance plan and the Risk Assessment Standardization Project Handbook. The SPAR model quality assurance plan provides reasonable assurance that the SPAR models used by agency risk analysts represent the as-built, as-operated plants to the extent intended within the scope of the SPAR models. As part of this plan, the staff periodically updates the SPAR models for operating NPPs to reflect the most recent operating experience and reliability data, performing routine updates to approximately 6 SPAR models per year. The Risk Assessment Standardization Project Handbook implements a formal, written process for maintaining SPAR models that are sufficiently representative of the as-built, as-operated plants to support model uses. The staff and Idaho National Laboratory also developed a SAPHIRE quality assurance program that is compliant with NUREG/BR-0167, “Software Quality Assurance Program and Guidelines,” and developed and released SAPHIRE Version 8, issued February 1993, which was independently verified and validated.

American Society of Mechanical Engineers and American Nuclear Society PRA Standard

In 2009, the staff, along with peer review teams comprised of industry experts, performed a peer review of a representative boiling-water reactor SPAR model and a representative pressurized-water reactor SPAR model in accordance with the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) PRA Standard, ASME RA-S-2002, “Standard for Probabilistic Risk Assessment for Nuclear Power Plant Applications,” and Regulatory Guide 1.200, “An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities.” The peer review teams concluded that—within constraints on access to licensee data and resources—the SPAR models are an appropriate tool to provide a check and to prompt questions on the licensee-maintained and peer reviewed PRA. The staff therefore concluded that SPAR models are an efficient tool for obtaining qualitative and quantitative insights for agency risk-informed applications.

Severe Accident Progression and Source Term Analysis

The MELCOR Code

The MELCOR code is a fully integrated, engineering-level computer code designed to model the progression of a broad spectrum of postulated severe accidents in light-water reactors and in nonreactor systems (e.g., spent fuel pool and dry cask). MELCOR has been under continuous development by the NRC and Sandia National Laboratories. Current activities involve the development and implementation of new and improved models to predict the severe accident behavior of various reactor (both light water and nonlight water) and spent fuel pool designs and to reduce modeling uncertainties. In addition, enhancements and more flexibility are being added to the code to evaluate the safety of accident-tolerant fuel designs. MELCOR represents the current state-of-the-art in accident progression analysis, which has developed from domestic and international research. The MELCOR code development meets the following criteria:

- The prediction of phenomena is in qualitative agreement with the current understanding of physics, and uncertainties are in quantitative agreement with experiments.

- The focus is on mechanistic models, where feasible, with adequate flexibility for parametric models.

- The code is portable, robust, and relatively fast running, and its maintenance follows established Software Quality Assurance standards.

- Detailed code documentation (including user guide, model reference, and assessment) is available.

The NRC uses MELCOR to model severe accident progression and to compute the resulting source terms for use in plant-specific PRAs and regulatory and backfit analyses. Recent examples include the technical bases for the following NRC studies:

- Enclosure H-3, "Summary of Detailed Analyses for SECY-12-0157," of this appendix summarizes the detailed analyses supporting SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012.

- Enclosure H-4, "Summary of Detailed Analyses for SECY-15-0085," of this appendix summarizes the detailed analyses supporting SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling-Water Reactors Rulemaking Activities," dated June 18, 2015; the NRC subsequently published the detailed analyses as NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling-Water Reactors with Mark I and Mark II Containments," issued March 2018.

- Enclosure H-5, "Summary of Detailed Analyses for SECY-13-0112 and NUREG-2161," of this appendix summarizes the detailed analyses supporting SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," dated October 9, 2013, which was documented in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," issued September 2014.

- Enclosure H-6, "Summary of Detailed Analyses in COMSECY-13-0030, Enclosure 1," of this appendix summarizes the detailed analyses supporting COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 12, 2013.

Level 1 success criteria analyses have used MELCOR, as noted in Figure H-11 (see, for example, NUREG/CR-7177, "Compendium of Analyses to Investigate Select Level 1

Probabilistic Risk Assessment End-State Definition and Success Criteria Modeling Issues," issued May 2014). The discussion of the MACCS code below notes a variety of NRC research studies that have used MELCOR. Additionally, some international organizations have used the code to assess severe accident management strategies.

MELCOR Code Structure

MELCOR is a modular code consisting of three general types of packages: (1) basic physical phenomena (i.e., hydrodynamics—control volume and flowpaths, heat and mass transfer to structures, gas combustion, and aerosol and vapor physics), (2) reactor-specific phenomena (i.e., decay heat generation, core degradation and relocation, ex-vessel [outside the reactor vessel] phenomena, and engineering safety systems), and (3) support functions (i.e., thermodynamics, equations of state, material properties, data-handling utilities, and equation solvers). These packages model the major systems of an NPP and their associated interactions. The various code packages have been written with well-defined interfaces between them. This allows the exchange of complete and consistent information among them so that all phenomena are coupled at every step.

MELCOR modeling makes use of a control volume approach in describing the plant system. No specific nodalization (how the control volumes are defined) of a system is forced on the user, which allows a choice of the degree of detail appropriate to the task at hand. Reactor-specific geometry is imposed only in modeling the reactor core. Even here, one basic model suffices for representing various core and fuel assembly designs, and a wide range of levels of modeling detail is possible.

MELCOR Source Term

The MELCOR output binary plot file contains the time-dependent variables of interest as a function of time at a frequency specified by the user. Of interest in Level 2 and Level 3 consequence analyses, MELCOR provides data on fluid flows and radionuclide transport to the environment through flowpaths identified as release paths. This information constitutes the source term and defines the magnitude and timing of the release of radionuclides. It is characterized by the following MELCOR plot variables:

- nominal aerosol density
- fluid temperature
- enthalpy
- cumulative fluid mass flow
- released radioactive mass for each radionuclide class
- aerosol size distribution

This information can be converted into a MACCS input file by the MelMACCS preprocessor code. The sections below describe MelMACCS, along with other associated codes.

ASME/ANS Level 2 PRA Standard

In January 2015, ASME/ANS issued for trial use “ASME/ANS RA-S-1.2-2014: Severe Accident Progression and Radiological Release (Level 2) PRA Standard for Nuclear Power Plant Applications for LWRs.” The NRC’s Site Level 3 PRA Level 2 analysis team used a prepublication draft of this trial use Level 2 PRA standard in a pilot application to perform a self-assessment of its draft internal events and floods Level 2 PRA.

Severe Accident Consequence Analysis

The MELCOR Accident Consequence Code System (MACCS)

MACCS is the NRC code used to estimate the offsite consequences associated with a hypothetical release of radioactive material into the atmosphere from a severe accident at an NPP. The code models atmospheric transport and dispersion (ATD); mitigative actions based on dose projections; dose accumulation by several pathways, including food and water ingestion; early and latent health effects; and economic costs. MACCS is currently the only code used in the United States for the offsite consequence analyses portion of NPP Level 3 PRAs.

As indicated in the main body of this NUREG, the NRC uses MACCS to estimate the averted offsite property damage cost and the averted offsite dose cost elements in the performance of cost-benefit analyses as part of backfit and regulatory analyses. The NRC has also used MACCS to support calculations of individual latent cancer fatality and prompt fatality risks for comparison to quantitative health objectives. As with the previous discussion on MELCOR, recent examples in which the NRC used MACCS in regulatory analyses include SECY-12-0157, SECY-15-0085, SECY-13-0112, and COMSECY-13-0030. The U.S. NPP license renewal applicants use MACCS to support the plant-specific evaluation of severe accident mitigation alternatives (SAMAs) that may be required as part of the applicant’s environmental report for license renewal. Additionally, MACCS is used in severe accident analyses and severe accident mitigation design alternative (SAMDA) assessments for environmental analyses supporting design certification, early site permit, and combined construction and operating license reviews for new reactors.

A variety of NRC research studies also used MACCS. The State-of-the-Art Reactor Consequence Analyses (SOARCA) project used MELCOR and MACCS to develop best estimates of the offsite radiological health consequences for potential severe reactor accidents at Peach Bottom Atomic Power Station (Peach Bottom), the Surry Power Station, and the Sequoyah Nuclear Plant. The MELCOR and MACCS best practices as applied in the 2012 SOARCA project were respectively documented in NUREG/CR-7008, “MELCOR Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses Project,” and NUREG/CR-7009, “MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses Project,” both issued August 2014. Three SOARCA uncertainty analyses have also been completed, including one for the Peach Bottom unmitigated long-term station blackout, documented in NUREG/CR-7155, “State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” issued May 2016. These studies propagated uncertainty for a variety of key uncertain MELCOR and MACCS parameters to develop insights into the overall sensitivity of SOARCA results and conclusions to input uncertainty and to identify the most influential input parameters for accident progression and offsite consequences. MACCS was also used in a consequence study of a beyond-design-basis earthquake affecting the spent fuel pool for a U.S. Mark I boiling-water reactor and is documented in NUREG-2161. In addition, the NRC's Full-Scope Site Level 3 PRA for a reference NPP site uses MACCS to support the offsite consequence analyses.

MACCS Code Structure

The MACCS code is subdivided into three modules that handle the various components of the consequence analysis calculation: ATMOS, EARLY, and CHRONC. These modules estimate consequences in sequential steps:

1. ATMOS models atmospheric transport and deposition of radioactive materials onto land and water bodies.

2. EARLY calculates the acute and lifetime doses, along with the associated health effects, during the emergency phase simulation.

3. CHRONC calculates the estimated exposures and health effects during an intermediate period of up to 1-year (intermediate phase) and computes the long-term (e.g., 50 years) exposures and health effects (late-phase model). CHRONC also calculates the economic costs of the intermediate and long-term protective actions, as well as the cost of the emergency response actions in the EARLY module.

The following sections summarize the MACCS code models. More detailed descriptions appear in the MACCS Code User Guide and Model Description, which includes NUREG/CR-4691, "MELCOR Accident Consequence Code System," issued February 1990 (NRC, 1990a) and NUREG/CR-6613, "Code Manual for MACCS2," issued May 1998 (NRC, 1998).

Atmospheric Transport and Dispersion

ATMOS models the dispersion of radioactive materials released into the atmosphere using the straight-line Gaussian plume segment model with provisions for meander and surface roughness effects. The ATD model treats buoyant plume rise, initial plume size caused by building wake effects, release of up to 500 plume segments, dispersion under given meteorological conditions, deposition under given dry and wet (precipitation) conditions, and decay and ingrowths of up to 150 radionuclides and a maximum of six generations.

The analyst has the option of using a single weather sequence. Sampling among multiple weather sequences is used in probabilistic consequence analysis studies to evaluate the variability in consequences that can result from uncertain weather conditions at the time of a future, hypothetical release of radioactive material. The results generated by the ATD model include radionuclide concentrations in air, on land, and as a function of time and distance from the release source; these results are subsequently used to model early, intermediate, and long-term phase radiological exposure, as discussed below.

Early (Emergency) Phase Protective Actions and Exposure Pathways

The EARLY module in MACCS assesses the time period immediately following a radioactive release while releases are ongoing. This is analogous to the emergency phase of a severe accident. Early phase exposure calculations account for reductions in dose from the use of emergency response measures such as sheltering, evacuation, and relocation of the population.

MACCS models sheltering and evacuation for user-specified population cohorts.[24] Different shielding factors for the different exposure pathways (i.e., cloudshine, groundshine, inhalation, and deposition on the skin) are associated with three types of activities: (1) normal activity, (2) sheltering, and (3) evacuation.

Intermediate Phase Protective Actions and Exposure Pathways

MACCS can model an intermediate phase following the end of the early phase. The only protective action modeled in this phase is relocation. If the projected dose to a population exceeds a user-specified threshold over a user-specified time duration, the population is assumed to be relocated to an uncontaminated area for the entire duration of this phase. The user defines a corresponding per-capita per diem economic cost. If the projected dose does not reach the user-specified threshold, MACCS models exposure pathways for groundshine and inhalation of resuspended material.

Long-Term Phase Protective Actions and Exposure Pathways

In the long-term phase, which follows the intermediate phase and can last, from months to years, protective actions are defined to keep the dose to an individual below specified limits. Protective actions in this phase include dose reduction measures, such as decontamination and interdiction of contaminated areas. Decisions on protective actions are based on two sets of independent criteria relating to whether land, at a specific location and time, is suitable for human habitation (habitability) or agricultural production (farmability). Habitability and farmability are defined by a set of user-specified maximum doses and a user-specified exposure period to receive those doses. The long-term phase includes both direct exposure pathways (i.e., groundshine, resuspension inhalation) and indirect exposure pathways through ingestion (i.e., food and water consumption).

Health Effects Modeling

MACCS employs a user-specified dose conversion factor file based on the most recent U.S. Environmental Protection Agency (EPA) guidance, currently, EPA's Federal Guidance Report No. 13, "Cancer Risk Coefficients for Environmental Exposure to Radionuclides," issued September 1999. Federal Guidance Report No. 13 converts the integrated air concentration and ground deposition of 825 radionuclides to a whole-body effective dose and individual organ doses for 26 tissues and organs and for four exposure pathways. In general, the radiological dose to a receptor (i.e., person) in each spatial element (i.e., an area of land) is the product of the radionuclide concentration or quantity, the exposure duration, the shielding factor, the dose conversion factor, and the usage factor (e.g., breathing rate). The total dose to an organ or the whole body is then obtained by summation across the relevant exposure pathways and radionuclides.

Offsite Consequence Measures

The results of a MACCS analysis can be reported in terms of population dose, health risks to the public, land contamination, population subject to long-term protective actions, and economic costs. Consequence results discussed in this section are conditional consequences (i.e., assuming the accident occurs). Therefore, this section does not consider the different

[24] Cohorts are subsets of the population with similar characteristics (e.g., school children in school at the time of the accident).

probabilities or frequencies of the different accident progression scenarios. Typical cost-benefit analyses and SAMDA/SAMA analyses generally report the individual risks, population dose, and economic costs as mean values (i.e., expected values). The values are averaged over sampled weather conditions representing a year of meteorological data and over the entire residential population within a circular or annular region. Past PRA applications have also shown complementary cumulative distribution functions of these consequence measures (the outputs of analysis), illustrating variability across weather conditions (inputs to the analysis).

Population Dose

As noted above, in general, the radiological dose to a receptor in each spatial element is the product of the radionuclide concentration or quantity, the exposure duration, the shielding factor, the dose conversion factor, and the usage factor (e.g., breathing rate). The total dose to an organ or the whole body is then obtained by summation across the relevant exposure pathways and radionuclides. Long-term population dose results are summed over the user-specified areas of interest and reported in person-Sieverts.

Individual (Population-Weighted) Latent Cancer Fatality Risk and Early Fatality Risk

The individual, population-weighted, latent cancer fatality[25] risk calculations include only the direct exposure pathways (i.e., groundshine, cloudshine, cloud inhalation, and resuspension inhalation) and exclude the ingestion (i.e., consumption of food and water) pathways. The MACCS early fatality model provides a pooled risk estimate of death from any of a number of competing causes of early death, such as hematopoietic, gastrointestinal, and pulmonary syndromes. Only the early phase exposure pathways are considered in the calculation of individual early fatality risk. The individual latent cancer fatality and early fatality risks are computed over user-specified regions. For example, for a large light-water reactor, a 10-mile radius circular region centered on the plant is used, for purposes of comparison to the latent cancer fatality risk quantitative health objective, and within 1 mile of the site boundary is used, for purposes of comparison to the prompt fatality risk quantitative health objective (NRC, 1986).

Economic Consequences

The offsite economic consequences model in MACCS estimates the direct offsite costs that result from protective actions modeled to reduce radiation exposures to the public. The current cost-based economic model treats the following costs:

- Evacuation costs: The daily cost of compensation for evacuees could include food, housing, transportation, and lost income.

- Relocation costs: The costs associated with relocating individuals during the intermediate and long-term phases.

- Decontamination of property: Costs are to decontaminate inhabited areas and farmland.

- Loss of use: Economic losses from loss of return on investment and depreciation of property value are incurred while property is temporarily interdicted. The depreciation of value of the buildings and other structures results from lack of habitation and maintenance.

[25] This is a fatal cancer incurred from radiological exposure.

- Condemnation of property: Economic losses result from the permanent interdiction of property.

- Disposal of contaminated farm products and interdiction of farming: The economic cost is from the loss of sales of farm products.

To obtain the total offsite economic costs, all the costs for the six cost categories are summed over the entire region of interest affected by the atmospheric release. Many of the values affecting the economic cost model are user inputs and thus can account for a variety of costs and can be adjusted for inflation, new technology, or changes in policy or practices.

Ongoing Updates

Work is ongoing to update the MACCS code to include additional state-of-practice modeling approaches (SECY-12-0110, "Consideration of Economic Consequences within the U.S. Nuclear Regulatory Commission's Regulatory Framework," Enclosure 9, "MELCOR Accident Consequence Code System, Version 2 (MACCS2)," dated August 14, 2012). Alternate ATD models are being implemented within MACCS by adding the capability to use results from the National Oceanic and Atmospheric Administration's HYbrid Single-Particle Lagrangian Integrated Trajectory (HYSPLIT) code (Stein et al., 2015). This will allow the use of models that may provide a better representation of atmospheric transport, dispersion, and deposition at longer ranges or in complex windfields. In addition, an alternative economic model will use regional gross domestic product-based input-output models to capture the upstream supply chain impacts of affected industries outside areas directly affected by radiological releases.

Associated Codes

WinMACCS

WinMACCS is a graphical user interface that assists the user in constructing and executing MACCS input files. The graphical user interface acts as a wizard that identifies what input is necessary for a particular calculation. WinMACCS allows the user to interact with graphical tools to aid in user input by visualization, such as defining an evacuation network using a map with the polar grid superimposed.

MelMACCS

MelMACCS is a graphical user interface that converts source term information from the severe accident analysis code MELCOR into a form suitable for use in the consequence analysis code MACCS. MelMACCS processes MELCOR information for use in the ATMOS package of MACCS for atmospheric transport and dispersion. Not all MACCS variables for source term input are directly obtained from a MELCOR plot file. The variables not provided are either calculated from other values in the plot file or are requested in the MelMACCS interface.

SecPop

SecPop is a preprocessor code for MACCS that enables the use of site-specific population, economic, and land use data in the calculation of offsite consequences. SecPop uses a block-level database of the U.S. population based on the U.S. Census and county-level data for economic information from the U.S. Department of Agriculture Census of Agriculture and Bureau of Economic Analysis. SecPop allows the user to scale population and economic data from the database years to a target year based on a user-specified growth rate. The output of SecPop is a site file that is input into MACCS. NUREG/CR-6525, Revision 2, “SecPop Version 4: Sector Population, Land Fraction, and Economic Estimation Program,” issued June 2019, provides more information.

COMIDA2

COMIDA2 is a preprocessor code that models the food-chain dose pathway. COMIDA2 can calculate estimates of radionuclide concentrations in agricultural products after a radioactive release following a hypothetical severe accident. This code calculates the uptake of radioisotopes into the edible portions of plants as a function of the development of the plant. It also considers the decay chains of nuclides, up to four daughters, and can, therefore, consider the loss and ingrowth of radioisotopes in the plant.

ENCLOSURE H-2: SUMMARY OF THE STATE-OF-THE-ART REACTOR CONSEQUENCE ANALYSES (SOARCA) PROJECT

Project Overview

The U.S. Nuclear Energy Commission (NRC) initiated the State-of-the-Art Reactor Consequence Analyses (SOARCA) project to further its understanding of the realistic consequences of severe reactor accidents. SOARCA addresses the consequences of rare but severe accidents at commercial reactors in the United States. The SOARCA analysts focused on accident progression, source term, and conditional consequences should the postulated accidents occur. The project did not include within its scope new work to calculate the frequencies associated with the postulated severe accidents.

The project, which began in 2006, combined information available at the time about the pilot plants' layout and operations, local population and site data, and emergency preparedness plans. The NRC analyzed information using the MELCOR and MELCOR Accident Consequence Code System (MACCS) suite of computer codes for integrated severe accident progression and offsite consequence modeling. The modeling incorporated insights from decades of research into severe reactor accidents.

Plants and Accident Scenarios Studied

The NRC staff initially evaluated potential consequences of select, important severe accidents at the Peach Bottom Atomic Power Station (Peach Bottom) and Surry Power Station (Surry) (NRC, 2012a). Selected accidents included station blackout scenarios for both plants and bypass scenarios for Surry. Peach Bottom is a General Electric boiling-water reactor with a Mark I containment, located in Pennsylvania; Surry is a Westinghouse 3-loop pressurized-water reactor (PWR) with a subatmospheric large, dry containment, located in Virginia. The staff subsequently evaluated a more limited set of scenarios at a third plant, the Sequoyah Nuclear Plant (Sequoyah), a Westinghouse 4-loop PWR with an ice condenser containment, located in Tennessee (NRC, 2019a). The Sequoyah study focused on issues unique to the ice condenser containment design because of its lower design pressure and smaller volume. For this third study, the staff also conducted an uncertainty analysis for one of the scenarios concurrently with the deterministic calculations, in which it conducted uncertainty analyses for one scenario each at the Peach Bottom and Surry plants after the initial deterministic SOARCA calculations (NRC, 2016b and NRC, 2015a, a draft that will be updated for the Surry uncertainty analysis).

The SOARCA project's main findings fall into three basic areas: how a reactor accident progresses, how existing systems and emergency measures can affect an accident's outcome, and how an accident would affect public health. The 2012 project findings, corroborated by subsequent uncertainty analyses and the Sequoyah analyses, include the following:

- Existing resources and procedures can stop an accident, slow it down, or reduce its impact before it can affect public health, if successfully implemented.

- Even if accidents proceed without successful intervention, they generally take longer to happen and release less radioactive material within the simulation time than earlier analyses suggested. Hence, some accidents that may have been traditionally classified as large-early release scenarios (e.g., interfacing systems loss-of-coolant accident for

Surry) may no longer contribute to large early release frequency because release is delayed beyond the time assumed to successfully evacuate the close-in population.

- The analyzed accidents pose “essentially zero” risk of early death (from radiological consequences) and only a negligible increase in the risk of a long-term cancer death, to a member of the public.

- The small risk for the calculated individual cancer fatalities is dominated by the long-term accumulation of very small doses (below allowable habitability criteria) to the public in the affected area.

The NRC makes supporting technical information available on the deterministic Peach Bottom analysis and Surry analysis in NUREG/CR-7110, “State-of-the-Art Reactor Consequence Analyses Project: Peach Bottom Integrated Analysis,” Volume 1, issued May 2013 (NRC, 2013a), and NUREG/CR-7110, “State-of-the-Art Reactor Consequence Analyses Project: Surry Integrated Analysis,” Volume 2, issued August 2013 (NRC, 2013b). NUREG/BR-0359, “Modeling Potential Reactor Accident Consequences,” issued December 2012, describes this Peach Bottom and Surry research for a general audience. The Peach Bottom uncertainty analysis of the unmitigated long-term station blackout (LTSBO) scenario is available in NUREG/CR-7155, “State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station,” issued May 2016. The Sequoyah integrated deterministic and uncertainty analyses are available in NUREG/CR-7245, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Sequoyah Integrated Deterministic and Uncertainty Analyses,” issued October 2019 (NRC, 2019a). The Surry uncertainty analysis of the unmitigated short-term station blackout (STSBO), including a potential induced steam generator tube rupture, is available in NUREG/CR-7262, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Project: Uncertainty Analysis of the Unmitigated Short-Term Station Blackout of Surry Power Station,” issued in 2020 (NRC, 2020).

Results of the Mitigated Scenarios

One of the goals of the original Peach Bottom and Surry SOARCA analyses was to study the benefits of the then-recently established mitigation measures in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh) (formerly B.5.b) for the accidents analyzed. All mitigated cases of SOARCA scenarios, except for one, result in prevention of core damage or no offsite release of radioactive material. The only mitigated case still leading to an offsite release was the Surry STSBO-induced steam generator tube rupture. In this case, mitigation is still beneficial in that it keeps most radioactive material inside containment and delays the onset of containment failure by about 2 days (NRC, 2012a). The NRC made no attempt to quantify the likelihood that mitigation would be successful and conducted no human reliability analysis. Instead, the scenarios were analyzed twice—one case assuming that mitigation was successful and an unmitigated case assuming successful mitigation did not occur.

The mitigated scenarios show zero individual early fatality risk from radiation exposure and zero risk or a very small risk of long-term cancer fatalities, depending on the specific scenario. The SOARCA results demonstrate the potential benefits of the mitigation measures analyzed in this project. SOARCA shows that successful mitigation either prevents core damage or prevents, delays, or reduces offsite health consequences.

The NRC was nearing completion of the SOARCA analyses when the accident at the Fukushima Dai-ichi plants in Japan occurred in 2011. The NRC did not redefine or reanalyze the scenarios following the Fukushima accident. It included a brief comparison to the Fukushima Dai-ichi nuclear power plant accident in the Peach Bottom uncertainty analysis technical report (NRC, 2016b). None of the SOARCA analyses included the use of flexible coping strategies (FLEX) because FLEX was still under development at the time of the analysis.

Results of Unmitigated Scenarios

Even the unmitigated scenarios result in essentially zero individual early fatality risk from radiation exposure. Although these unmitigated scenarios result in core damage and release of radioactive material to the environment, the release is delayed, which allows the population to take protective actions (including evacuation and sheltering). The individual risk of long-term cancer fatality is calculated to be very small. Table H-8 shows the point estimates (NRC, 2012a; NRC, 2019a), as well as uncertainty analysis bands where available (NRC, 2016b; NRC, 2019a; NRC, 2020), for the conditional risk (assuming that the accident occurs) to the public living between 0 and 10 miles from the plants, assuming the linear no-threshold dose response model. The SOARCA analyses calculated risk to individuals out to 50 miles from the plants. For some scenarios, the risks to the 10- to 30-mile population (outside the plume exposure pathway emergency planning zone) are slightly higher than the risk to the 0- to 10-mile population. Considering that the frequencies estimated for these scenarios are in the range of one per 100,000 to one per 30 million reactor-years, the absolute risk of long-term cancer fatality from the analyzed SOARCA scenarios is projected to be negligible.

Table H-8 Conditional Annual Average Individual Latent Cancer Fatality Risk from SOARCA Unmitigated Scenarios within 10 miles of the Plant

Scenario	Peach Bottom		Surry				Sequoyah
	LTSBO	STSBO	LTSBO	STSBO	Induced SGTR	ISLOCA	STSBO
Point estimate[a]	9x10^{-5}	2x10^{-4}	5x10^{-5}	9x10^{-5}	3x10^{-4}	3x10^{-4}	8x10^{-5}
5th percentile[b]	3x10^{-5}	N/A	N/A	3x10^{-7}	N/A	N/A	1x10^{-8}
95th percentile[b]	4x10^{-4}			2x10^{-4}			2x10^{-4}

[a] The Peach Bottom and Surry accident simulations were carried out to 48 hours; the Sequoyah accident was simulated out to 72 hours.
[b] The Peach Bottom uncertainty analysis simulation was carried out to 48 hours; the Surry and Sequoyah uncertainty analysis simulations were carried out to 72 hours. The Surry STSBO 5th and 95th percentiles include induced steam generator tube rupture (SGTR).

Notable Assumptions

The SOARCA models assume that 99.5 percent of the population residing in the 10-mile emergency planning zone will evacuate as ordered. Shadow evacuations—the voluntary evacuation of members of the public who have not been ordered to evacuate—are also modeled for 10- to 15-mile or 10- to 20-mile radius annular rings around the plants. The Sequoyah analysis explicitly considered the potential impact of the seismic initiating event on emergency response and included sensitivity calculations for extended sheltering-in-place with and without degraded shielding caused due to structural damage, in case evacuation is delayed (NRC, 2019a). The Peach Bottom and Surry calculations assume the unmitigated accident releases can be terminated within 48 hours. The Sequoyah calculation assumes releases can be terminated within 72 hours.

Uses of SOARCA Models and Insights

SOARCA models and insights were subsequently leveraged in a variety of projects, including the analyses summarized in Enclosures H-3 through H-6 to this appendix. The NRC also published research Information Letter 19-01, “Benefits and Uses of the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project,” issued 2019 (NRC, 2019c), which summarizes many of the uses of the SOARCA body of work.

ENCLOSURE H-3: SUMMARY OF DETAILED ANALYSES FOR SECY-12-0157, "CONSIDERATION OF ADDITIONAL REQUIREMENTS FOR CONTAINMENT VENTING SYSTEMS FOR BOILING WATER REACTORS WITH MARK I AND MARK II CONTAINMENTS"

This enclosure summarizes the 2012 analyses supporting the consideration of additional requirements for containment venting systems for boiling-water reactors (BWRs) with Mark I and Mark II containments, following the 2011 accident at the Fukushima Dai-ichi nuclear power plant in Japan. The contents of this enclosure should be considered with the Commission direction in its staff requirements memorandum (SRM)–SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated March 19, 2013, and the subsequent analysis described in Enclosure H-4, "Summary of Detailed Analyses for SECY-15-0085, 'Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling-Water Reactors Rulemaking Activities'" to this appendix. A summary of SRM-SECY-12-0157 is provided at the end of this enclosure.

Problem Statement and Regulatory Objectives

The accident that occurred on March 11, 2011, at the Fukushima Dai-ichi nuclear power plant in Japan underscored the potential need for nuclear power plant safety improvements related to beyond-design-basis events involving natural hazards and their causal effects on plant systems and barriers from an extended loss of electrical power and access to heat removal systems. As part of its response to lessons learned from this accident, the U.S. Nuclear Regulatory Commission (NRC) staff issued Order EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," dated March 12, 2012." This order required licensees that use the boiling-water reactor (BWR) with Mark I and Mark II containment designs to install hardened containment vents. These hardened containment vents would address problems encountered during the Fukushima accident by providing plant operators with improved methods for venting containment during accident conditions and thereby preventing containment overpressurization and subsequent failure.

While developing the requirements for Order EA-12-050, the staff acknowledged that questions remained about maintaining containment integrity and limiting the release of radiological materials if licensees used the venting systems during severe accident conditions. In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff also identified the addition of an engineered filtered vent system to improve reliability and limit the release of radiological materials should the venting systems be used after significant core damage had occurred.

Regulatory Alternatives

The NRC considered four regulatory alternatives that address containment venting systems for BWRs with Mark I and Mark II containments in the regulatory analysis performed in support of SECY-12-0157:

- Option 1: Reliable Hardened Vents (Status Quo). Continue to implement Order EA-12-050 and install reliable hardened vents to reduce the probability of failure of BWR Mark I and Mark II containments and take no additional action to improve their ability to operate under severe accident conditions or to require the installation of an engineered filtered vent system. This alternative represented the status quo and served as the regulatory baseline against which the costs and benefits of other alternatives were measured.

- Option 2: Severe-Accident-Capable Venting System Order (without Filter). Upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions. This alternative would increase confidence in maintaining containment functionality following core damage events. Although venting containment during severe accident conditions may result in significant radiological releases, it would prevent overpressurization and reduce the probability of gross containment failures that could hamper accident management and result in larger radiological releases.

- Option 3: Filtered Severe Accident Venting System Order. Design and install an engineered filtered containment venting system that is intended to prevent the release of significant amounts of radiological materials for dominant severe accident scenarios at BWRs with Mark I and Mark II containments. The engineered filtering system would need to operate under severe accident conditions to reduce the amount of radiological material released to the environment from venting containment to prevent overpressurization.

- Option 4: Severe Accident Confinement Strategies. Pursue development of requirements and technical acceptance criteria for confinement strategies and require licensees to justify operator actions and systems or combinations of systems (e.g., suppression pools, containment sprays, and engineered filters) to accomplish the function and meet the requirements. For this option, the staff did not evaluate a specific filtering system; instead, it drew on insights from various sensitivity studies to define a possible approach.

Safety Goal Evaluation

This regulatory analysis required a safety goal evaluation because each of the alternatives was considered a generic safety enhancement backfit subject to the substantial additional protection standard in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.109 (a)(3). Each alternative, if implemented, would improve containment performance by reducing the probability of containment failure, given the assumed occurrence of a severe accident scenario, or the amount of radiological material released to the environment from a severe accident scenario, or both. However, since none of the alternatives would impact the frequency of core damage accidents (i.e., the change in core damage frequency (CDF) for each alternative relative to the regulatory baseline was zero), the safety goal screening criteria in the regulatory analysis guidelines could not be used to determine whether each alternative could result in a substantial increase in overall protection of public health and safety.

Therefore, the Japan Lessons-Learned Steering Committee (NRC, 2011c) evaluated whether imposition of requirements for severe-accident-capable or filtered venting systems would satisfy the substantial additional protection standard. The Japan Lessons-Learned Steering Committee decided that the staff should take the next step within the regulatory analysis process by estimating and evaluating the costs and benefits.

Technical Evaluation

To support the assessment of the quantitative costs and benefits of severe-accident-capable vents (Option 2) and filtered containment venting (Option 3), the staff (with support from Sandia National Laboratories) analyzed selected accident scenarios for a BWR plant with a Mark I containment. The staff used the NRC's severe accident analysis code, MELCOR, and the MELCOR Accident Consequence Code System (MACCS) to perform the analysis. The staff used the MELCOR code to calculate fission product release estimates for each of the selected accident scenarios, and this information was used in MACCS to calculate the offsite radiological consequences for each of the selected accident scenarios. Enclosure H-1, "Description of Analytical Tools and Capabilities," to this appendix describes these codes and their capabilities in more detail.

Accident Scenario Selection

The selection of accident scenarios considered for MELCOR and MACCS analyses was informed by both the State-of-the-Art Reactor Consequence Analyses (SOARCA) studies and a study of the Fukushima accident that Sandia National Laboratories was performing at the time. Two of the accident scenarios from the SOARCA study for Peach Bottom Atomic Power Station (Peach Bottom) selected for MELCOR and MACCS analyses were (1) the long-term station blackout (LTSBO) and (2) the short-term station blackout (STSBO).

MELCOR Severe Accident Progression and Source Term Analyses

Thirty MELCOR cases were run, simulating accident scenarios with different possible outcomes. Cases 2, 3, 6, 7, 12, 13, 14, and 15 became MELCOR base cases, with the results used for MACCS consequence calculations and for the regulatory analysis. The remaining cases were run as variations of the base cases for sensitivity analyses. The base cases represented the following accident scenarios:

- Case 2: No venting or spray

- Case 3: Wetwell venting but no spray

- Case 6: Core spray only

- Case 7: Core spray with wetwell venting

- Case 12: Drywell venting

- Case 13: Drywell venting and drywell spray

- Case 14: Drywell spray only

- Case 15: Drywell spray with wetwell venting

Collectively, the base cases encompassed all representative combinations of prevention and mitigation measures considered in the description of alternatives used in the regulatory analysis. Case 2 with no venting or spray mapped to Option 1 (status quo). Likewise, all venting cases (Cases 3, 7, 12, 13, and 15) mapped to Option 2 (severe-accident-capable vent) and—when considered in combination with an external filter—to Option 3 (filtered vent). Case 6 and Case 14 (both without venting but with sprays) were considered variations of Option 1.

The selected MELCOR accident scenarios were organized into four groups to compare the effect of venting and additional mitigation actions:

- Base case: Case 2 and Case 3

- Core spray after reactor pressure vessel failure: Case 6 and Case 7

- Main steamline failure with drywell venting at 24 hours: Case 12 and Case 13

- Drywell spray at 24 hours: Case 14 and Case 15

MACCS Consequence Analyses

The analysts used MACCS to perform consequence analyses for selected accident scenarios to calculate offsite doses and land contamination and their effect on members of the public with respect to individual prompt and latent cancer fatality risk, land contamination areas, population dose, and economic costs. They used the Peach Bottom unmitigated LTSBO MACCS input deck from the SOARCA study, with two key modifications. One modification was the modeling of the ingestion pathway, which was excluded in the SOARCA analyses. Another modification was the use of revised source terms calculated from the MELCOR analyses for this study to account for variation in the LTSBO scenario and the effect of adding an external filter to the vent paths.

Risk Evaluation

The analysts constructed a simplified event tree to estimate the radiological release frequencies of the MELCOR accident scenarios. Coupled with the MACCS consequence results developed for each MELCOR scenario, this simplified event tree provided the information needed to assess the reduction in risk resulting from the installation of a severe-accident-capable venting system. The simplified event tree structure used to estimate radiological release frequencies was designed to allow assessment of a wide range of severe-accident-capable vent system designs that varied depending on (1) where the vent is attached (wetwell or drywell), (2) how the vent is actuated (manually by the operator or passively using a rupture disk), and (3) whether the severe-accident-capable venting system has a filter. Table H-9 identifies the nine hypothetical plant modifications (“Mods”) that were assessed using the simplified event tree structure.

Table H-9 Hypothetical Plant Modifications

Identifier	Severe-Accident-Capable Vent Filter	Severe-Accident-Capable Vent Location	Severe-Accident-Capable Vent Actuation
Mod 0 (current situation)	NA	None	NA
Mod 1	No	Wetwell	Manual
Mod 2	No	Wetwell	Passive
Mod 3	No	Drywell	Manual
Mod 4	No	Drywell	Passive
Mod 5	Yes	Wetwell	Manual
Mod 6	Yes	Wetwell	Passive
Mod 7	Yes	Drywell	Manual
Mod 8	Yes	Drywell	Passive

The simplified event tree shown in Figure H-13 traced the accident progression starting from the onset of core damage. The first two event tree headings parsed the total CDF according to the type of hazard that initiated the accident (internal or external) and the type of core damage sequence (station blackout [SBO] sequences, bypass sequences in which venting containment has little or no impact because the containment is bypassed, fast sequences that evolve rapidly and reduce the available time for the operator to manually open the severe-accident-capable vent, and other sequences). Subsequent event tree headings consider (1) operation of the severe-accident-capable vent, (2) offsite power recovery (which is influenced by the type of hazard that initiated the accident), and (3) the availability of a water supply (portable pump) to the drywell. Each sequence was assigned to one of four possible containment status end states:

- Vented: The severe-accident-capable vent is opened, preventing containment overpressurization failure. A source of water to the drywell exists, preventing liner melt-through.

- Liner Melt-through (LMT): The severe-accident-capable vent is opened, preventing containment overpressurization failure. No source of water to the drywell exists, and liner melt-through occurs.

- Overpressurization (OP): The severe-accident-capable vent is closed, resulting in containment overpressurization failure. A source of water to the drywell exists, preventing liner melt-through.

- OP + LMT: The severe-accident-capable vent is closed, resulting in containment overpressurization failure. No source of water to the drywell exists, and liner melt-through occurs.

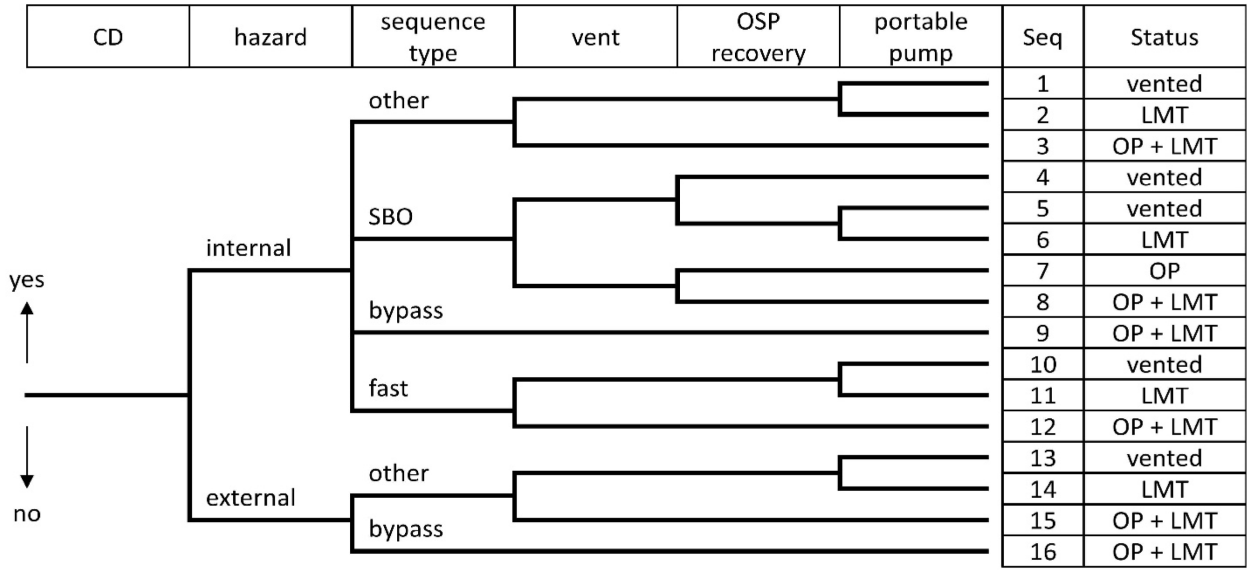

Figure H-13 Simplified Event Tree Structure

This simplified event tree delineates 16 post-core-damage accident sequences. Each sequence in the simplified event tree was assigned to a unique containment status. This mapping, together with the definitions of the hypothetical plant modifications shown in Table H-10, determined the specific MELCOR case and MACCS calculation that applies to each sequence, as shown in Table H-11.

Table H-10 Mapping of Simplified Event Tree Sequences to Plant Modifications and MELCOR Cases

Modification Description				Release Sequence Containment Status End State			
Mod	**Filter**	**Location**	**Actuation**	**Vented** Sequence: 1, 4, 5, 10, and 13	**LMT** Sequence: 2, 6, 11, and 14	**OP** Sequence: 7	**OP + LMT** Sequence: 3, 8, 9, 12, 15, and 16
0	NA	NA	None	NA	NA	Case 6	Case 2
1	No	Wetwell	Manual	Case 7 or 15 (no filter)	Case 3 (no filter)	Case 6	Case 2
2	No	Wetwell	Passive				
3	No	Drywell	Manual	Case 13 (no filter)	Case 12 (no filter)	Case 14	Case 2
4	No	Drywell	Passive				
5	Yes	Wetwell	Manual	Case 7 or 15 (filter)	Case 3 (filter)	Case 6	Case 2
6	Yes	Wetwell	Passive				
7	Yes	Drywell	Manual	Case 13 (filter)	Case 12 (filter)	Case 14	Case 2
8	Yes	Drywell	Passive				

Analysts developed parameter values based on information from a variety of sources to estimate the radiological release frequencies for each sequence in the simplified event tree. Table H-11 summarizes this information.

Table H-11 Parameter Values Used to Estimate Radiological Release Frequencies

Parameter	Value		Basis
CDF	2.0×10^{-5} per reactor-year (ry)		Standardized Plant Analysis Risk (SPAR) external hazard models
Fraction of total CDF due to external hazards	0.8		SPAR external hazard models; review of previous probabilistic risk assessments (PRAs)
Breakdown of sequence types for internal hazards[a]	Other	0.83	SPAR internal hazard models
	SBO	0.12	
	Bypass	0.05	
	Fast	0.01	
Breakdown of sequence types for external hazards[a]	Other	0.95	Review of previous PRAs; engineering judgment
	Bypass	0.05	
Probability that severe-accident-capable vent fails to open	Mod 0	1	Vent not installed
	Mods 1, 3, 5, 7—other or SBO	0.3	SPAR-H method (manual vent; longer available time)
	Mods 1, 3, 5, 7—fast	0.5	SPAR-H method (manual vent; shorter available time)
	Mods 2, 4, 6, 8	0.001	Engineering judgment (passive vent mechanical failure)
Conditional probability that offsite power is not recovered by the time of lower head failure given not recovered at the time of core damage (internal hazards)	0.38		Historical data (NUREG/CR-6890)
Probability that portable pump for core spray or drywell spray fails	0.3		SPAR-H; consistent with SPAR B.5.b study by Idaho National Laboratory

[a] The values may not total to one due to rounding.

MACCS is used to calculate the mean conditional offsite radiological consequences per release, conditioned on the assumed occurrence of the accident scenario that each MELCOR case represented. Table H-12 provides the mean results for the 50-mile population dose and 50-mile offsite cost consequence metrics.

Table H-12 Mean MACCS Consequence Results for Selected MELCOR Accident Scenarios

Case[a,b]	Core Spray	Drywell Spray	Venting	Location	50-mile Population Dose (person-rem/event)	50-mile Offsite Cost (million $/event)
2	no	no	no	NA	514,000	1,910
3F	no	no	yes	wetwell	183,000	274
3NF	no	no	yes	wetwell	397,000	1,730
6	yes	no	no	NA	305,000	847
7F	yes	no	yes	wetwell	37,300	18
7NF	yes	no	yes	wetwell	235,000	484
12F	no	no	yes	drywell	232,000	391
12NF	no	no	yes	drywell	3,810,000	33,300
13F	no	yes	yes	drywell	59,990	38
13NF	no	yes	yes	drywell	3,860,000	33,000
14	no	yes	no	NA	86,100	116
15F	no	yes	yes	wetwell	43,300	20
15NF	no	yes	yes	wetwell	280,000	588

[a] F: filtered case
[b] NF: not filtered case

The analysts calculated risk by combining the frequencies of radiological releases with their conditional offsite radiological consequences. Table H-13 provides the point estimate values for the 50-mile population dose risk and the 50-mile offsite cost risk for each of the nine hypothetical plant modifications.

Table H-13 Point Estimate Risk Values for Each Hypothetical Plant Modification

Mod	Vent Filtered	Vent Location	Vent Actuation	50-mile Population Dose Risk (person-rem/reactor-year [ry])	50-mile Offsite Cost Risk ($/ry)
0	NA	None	NA	10.2	$37,884
1	No	Wetwell	Manual	7.2	$24,041
2	No	Wetwell	Passive	5.9	$18,117
3	No	Drywell	Manual	54.5	$452,466
4	No	Drywell	Passive	73.5	$630,000
5	Yes	Wetwell	Manual	4.5	$13,958
6	Yes	Wetwell	Passive	2.0	$3,717
7	Yes	Drywell	Manual	4.9	$14,540
8	Yes	Drywell	Passive	2.6	$4,642

Table H-14 provides the risk reductions (relative to Mod 0, the current situation) associated with implementing plant modifications for the severe-accident-capable venting system (Mods 1 through 8). Figures H-14 and H-15 graphically illustrate this information.

Table H-14 Risk Reductions from Severe-Accident-Capable Venting System Plant Modifications

Mod	Vent Filtered	Vent Location	Vent Actuation	Reduction in 50-mile Population Dose Risk (Δperson-rem/ry)	Reduction in 50-mile Offsite Cost Risk (Δ$/ry)
1	No	Wetwell	Manual	3.0	$13,842
2	No	Wetwell	Passive	4.3	$19,767
3	No	Drywell	Manual	(44.3)[a]	($414,582)
4	No	Drywell	Passive	(63.3)	($592,117)
5	Yes	Wetwell	Manual	5.7	$23,926
6	Yes	Wetwell	Passive	8.2	$34,166
7	Yes	Drywell	Manual	5.3	$23,344
8	Yes	Drywell	Passive	7.6	$33,242

[a] Negative values are shown using parentheses (e.g., negative 44.3 is displayed as (44.3)).

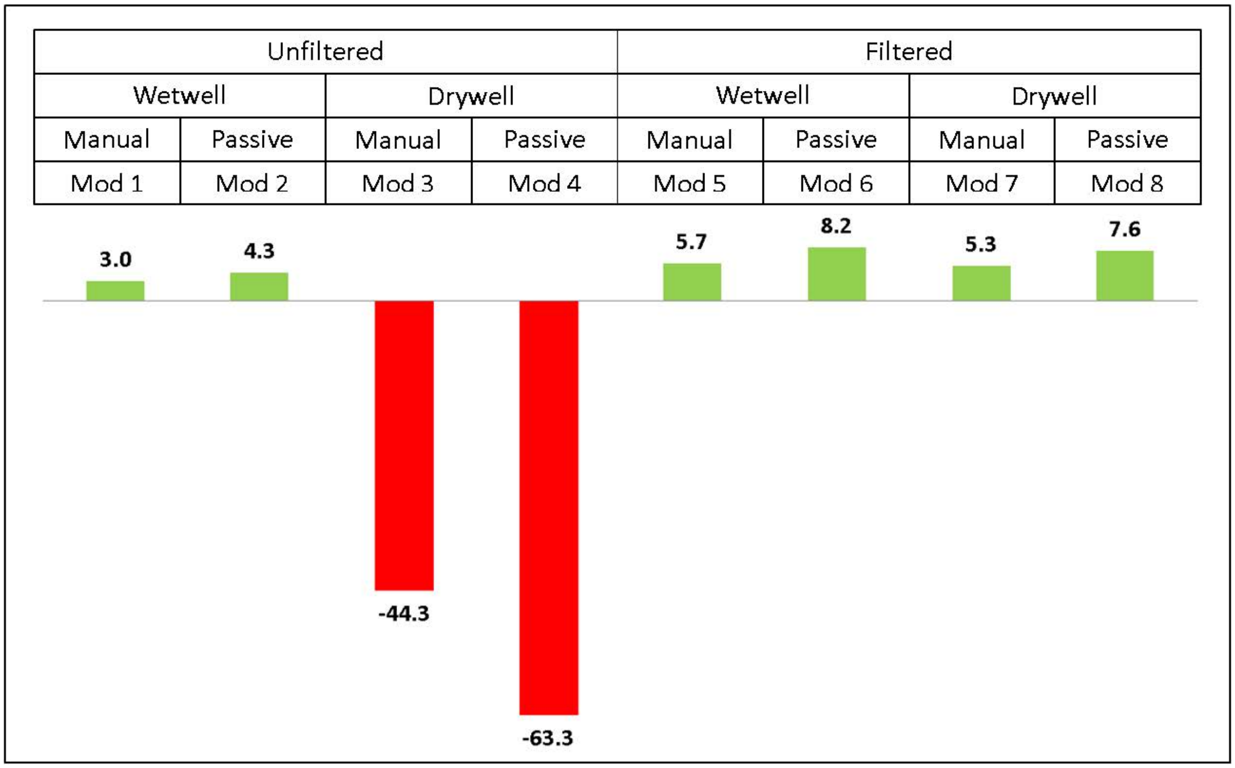

Figure H-14 Reduction in 50-mile Population Dose Risk (Δperson-rem/ry)

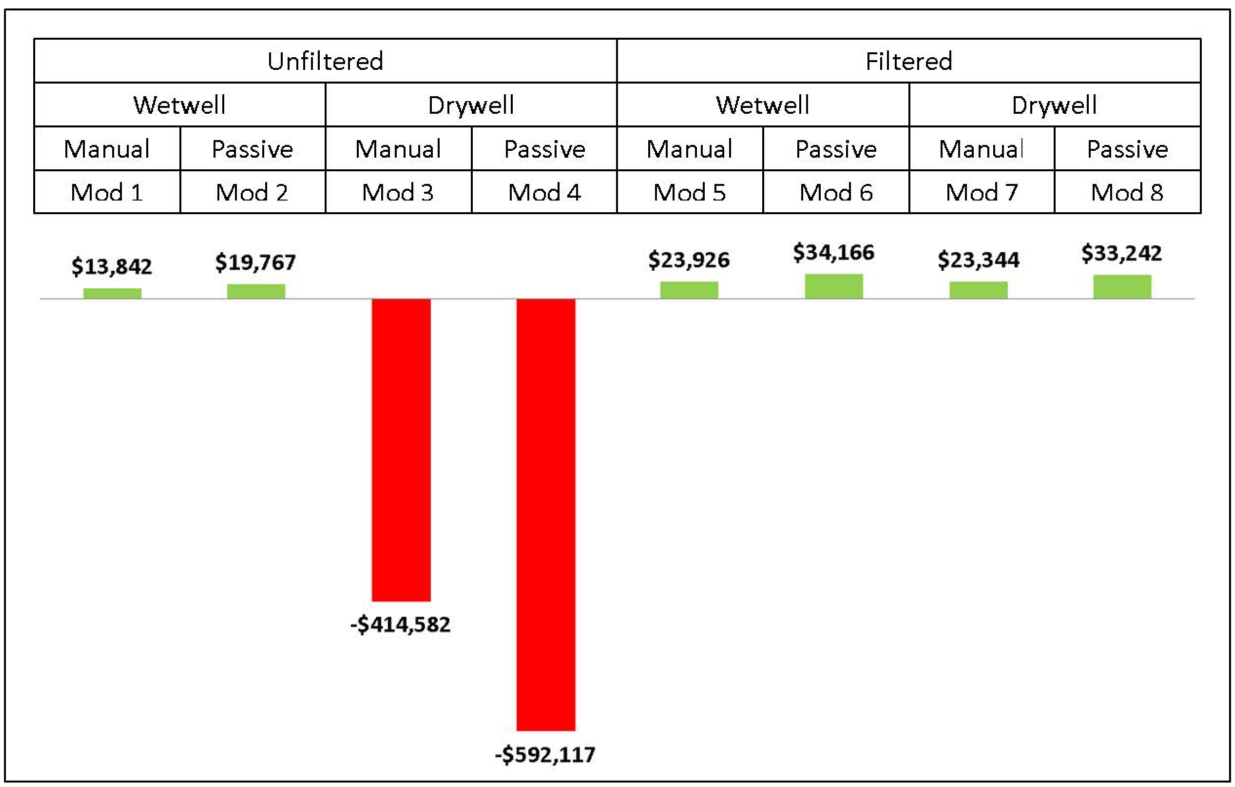

Figure H-15 Reduction in 50-mile Offsite Cost Risk (Δ$/ry)

To gain further insight into the risk reductions afforded by the hypothetical plant modifications, analysts performed a simple parametric Monte Carlo uncertainty analysis. They assigned an uncertainty distribution to each of the parameters used to quantify the radiological release frequencies and to each of the consequences. Table H-15 shows parameters that specify the uncertainty distribution.

Table H-15 Parameter Uncertainty Distributions

Parameter	Mean		Distribution
CDF	2.0×10^{-05}/ry		Lognormal; error factor = 10
Fraction of total CDF due to external hazards	0.8		Beta; α = 0.5, β = 0.125
Breakdown of sequence types for internal hazards	Other	0.83	Dirichlet[a] α1 (other) = 41 α2 (SBO) = 6 α3 (bypass) = 2.5 α4 (fast) = 0.5
	SBO	0.12	
	Bypass	0.05	
	Fast	0.01	
Breakdown of sequence types for external hazards	Other	0.95	Beta; α (bypass) = 0.5, β (bypass) = 9.5
	Bypass	0.05	
Probability that severe-accident-capable vent fails to open	Mod 0	1	Held constant
	Mods 1, 3, 5, 7—other or SBO	0.3	Beta; α = 0.5, β = 1.167
	Mods 1, 3, 5, 7—fast	0.5	Beta; α = 0.5, β = 0.5
	Mods 2, 4, 6, 8	0.001	Beta; α = 0.5, β = 499.5
Conditional probability that offsite power is not recovered by the time of lower head failure given not recovered at the time of core damage (internal hazards)	0.38		Beta; α = 0.5, β = 0.816
Probability that portable pump for core spray or drywell spray fails	0.3		Beta; α = 0.5, β = 1.167
Consequences	Per Table H-6		Lognormal; error factor = 10 Within a given consequence category, consequences were assumed to be totally dependent.

[a] The Dirichlet distribution is a family of continuous multivariate probability distributions parameterized by a vector α of positive reals. It is a multivariate generalization of the Beta distribution. Dirichlet distributions are commonly used as prior distributions in Bayesian statistics.

Figures H-16 and H-17 show the results[26] of the parametric uncertainty analysis. These figures show that, although somewhat higher, the mean values are very close to the corresponding point estimates. In general, the ratio of the 95th percentile to the point estimate varies from 3.5 to 4.0 depending on the consequence category. The major contributors to uncertainty in the risk reduction results were uncertainty in both the CDF and the conditional consequences.

[26] These figures do not show the results of Mods 3 and 4 because the results are negative (i.e., detrimental compared to the status quo), as shown in Figures H-16 and H-17.

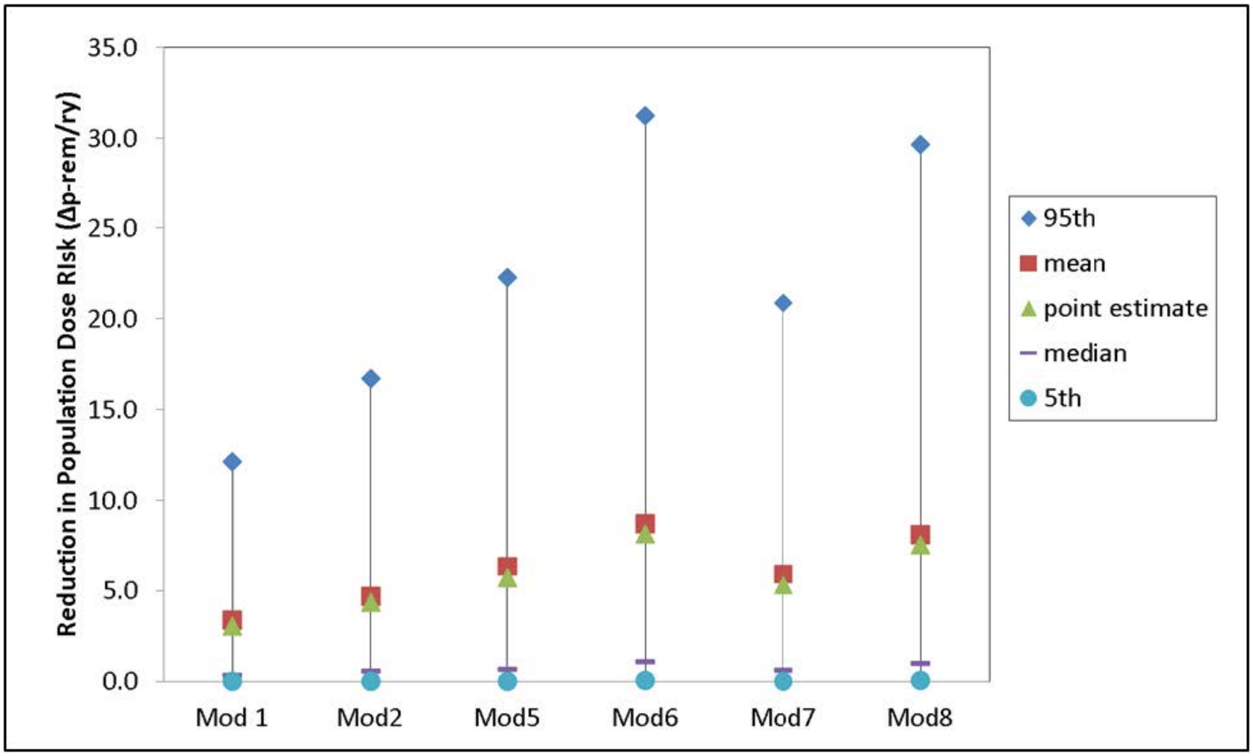

Figure H-16 Uncertainty in Reduction in 50-mile Population Dose Risk

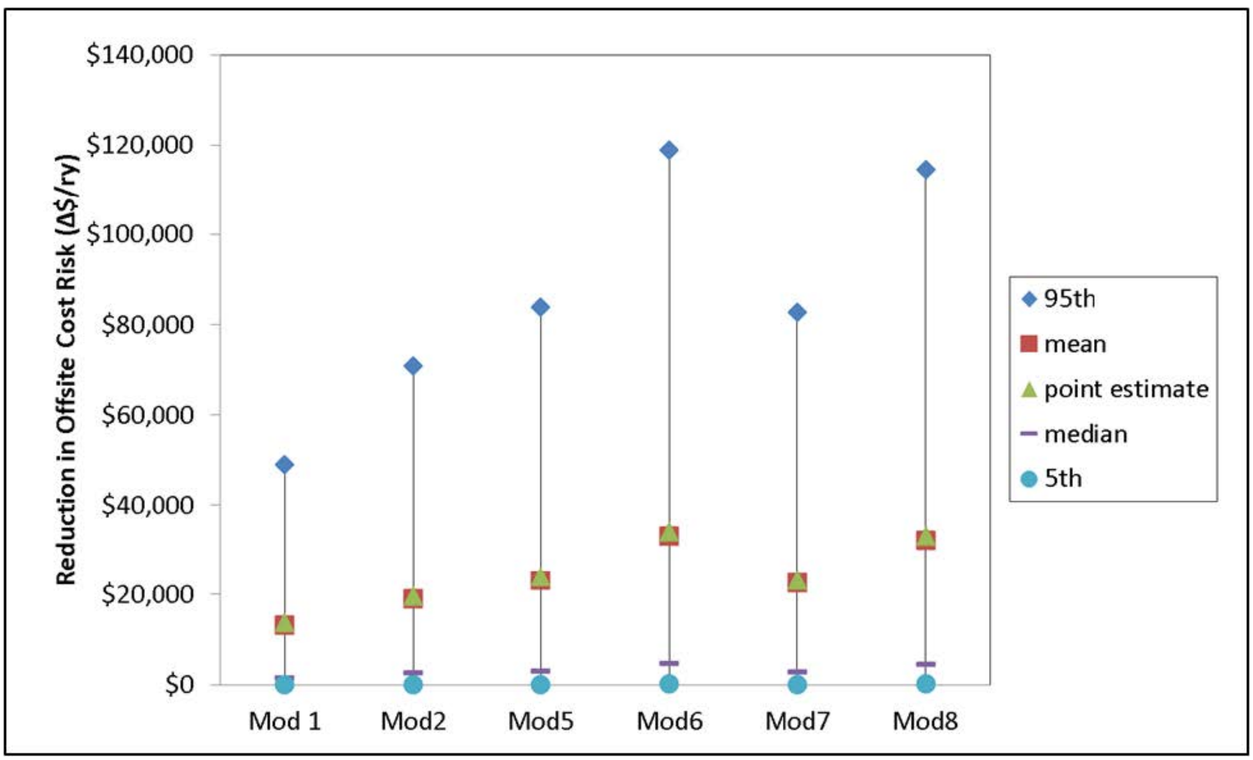

Figure H-17 Uncertainty in Reduction in 50-mile Offsite Cost Risk

These risk results that incorporated insights from the MELCOR and MACCS analyses led to the following specific conclusions about severe-accident-capable venting:

- The installation of an unfiltered wetwell severe-accident-capable venting system would reduce public health risk and offsite economic cost risk. By contrast, the installation of an unfiltered drywell severe-accident-capable venting system would increase public health risk and offsite economic cost risk.

- The installation of a filtered severe-accident-capable venting system (attached to either the wetwell or the drywell) would reduce public health risk and offsite economic cost risk. The installation of an external filter into the severe-accident-capable venting system is preferable.

- By preventing containment overpressurization failure, the successful operation of a severe-accident-capable venting system promotes access to plant areas where portable pumps could be installed to provide core debris cooling.

- Passive actuation (via a rupture disk) is preferred to manual actuation because it is more reliable and thus results in larger risk reductions.

- The uncertainty in the amount of risk reduction achieved by the installation of a severe-accident-capable venting system comes mainly from uncertainty both in the CDF and in the consequences resulting from radiological releases.

Cost-Benefit Analysis Results

The reductions in 50-mile population dose risk and 50-mile offsite cost risk (relative to Mod 0, the current situation) associated with implementation of the severe-accident-capable venting system plant modifications (Mods 1 through 8) were respectively used to calculate the values of the public health and offsite property attributes for Options 2 and 3 in a cost-benefit analysis. For the purposes of this analysis, Option 2 used the results for Mod 2 and Option 3 used the results for Mod 6. These results corresponded to the plant design modifications that achieved the largest risk reduction for each alternative.

Table H-16 summarizes the results of the quantitative cost-benefit analysis of a severe-accident-capable (Option 2) and filtered vent system (Option 3) that used the regulatory analysis guidelines that were in effect at the time. This table includes results for both the base-case analysis that used the best estimate CDF value of 2.0×10^{-5} per reactor-year and a one-way sensitivity analysis in which a CDF value of 2.0×10^{-4} per reactor-year was used to evaluate the impact on the results of varying this important uncertain parameter.

Table H-16 Summary of Quantitative Cost-Benefit Analysis Results for Filtered Containment Vent System using a $2,000 per Person-Rem Conversion Factor

Attribute	Severe-Accident-Capable Venting Systems		Engineered Filtered Venting Systems	
	Base Case[a] CDF=2.0x10^{-5}/ry	Sensitivity Case[a] CDF=2.0x10^{-4}/ry	Base Case[a] CDF=2.0x10^{-5}/ry	Sensitivity Case[a] CDF=2.0x10^{-4}/ry
Public Health	150	1,500	290	2,900
Occupational Health	11	110	19	190
Offsite Property	348	3,480	600	6,000
Onsite Property	268	2,680	430	4,300
Industry Implementation	(2,000)[b]	(2,000)	(15,000)	(15,000)
Industry Operation	n/a	n/a	(1,100)	(1,100)
NRC Implementation	(27)	(27)	(27)	(27)
Net Benefit	(1,250)	5,743	(14,778)	(2,737)

[a] Values are in thousand dollars per unit.
[b] Negative values are shown using parentheses (e.g., negative 2,000 is displayed as (2,000)).
(Source: SECY-12-0157, Enclosure 1, Table 1)

At the time of the analysis, the staff was updating the dollar per person-rem conversion factor policy and performed sensitivity analyses to evaluate the impact on results of increasing the dollar per person-rem conversion factor from $2,000 per person-rem to $4,000 per person-rem. Table H-17 summarizes the results of these sensitivity analyses.

Table H-17 Summary of Adjusted Quantitative Cost-Benefit Analysis Results for Filtered Containment Vent System using a $4,000 per Person-Rem Conversion Factor

Attribute	Severe-Accident-Capable Venting Systems		Engineered Filtered Venting Systems	
	Base Case[a] CDF=2.0x10^{-5}/ry	Sensitivity Case[a] CDF=2.0x10^{-4}/ry	Base Case[a] CDF=2.0x10^{-5}/ry	Sensitivity Case[a] CDF=2.0x10^{-4}/ry
Public Health	300	3,000	580	5,800
Occupational Health	22	220	38	380
Offsite Property	348	3,480	600	6,000
Onsite Property	268	2,680	430	4,300
Industry Implementation	(2,000)[b]	(2,000)	(15,000)	(15,000)
Industry Operation	n/a	n/a	(1,100)	(1,100)
NRC Implementation	(27)	(27)	(27)	(27)
Net Benefit	(1,089)	7,353	(14,479)	353

[a] Values are in thousand dollars per unit.
[b] Negative values are shown using parentheses (e.g., negative 2,000 is displayed as (2,000)).
(Source: SECY-12-0157, Enclosure 1, Table 3)

Qualitative Factors

Because the net benefits for both Option 2 and Option 3 were negative for the base case, the quantitative cost-benefit analysis did not appear to justify the imposition of additional requirements on the venting systems for BWR Mark I and Mark II containments under base-case assumptions. However, a one-way sensitivity analysis using a CDF value in the upper range of its uncertainty band resulted in a positive net benefit for Option 2, indicating it may be cost-beneficial. Moreover, a two-way sensitivity analysis within which the higher CDF value and a $4,000 per person-rem conversion factor was used resulted in a positive net benefit for both Option 2 and Option 3, indicating that both options may be cost-beneficial, with Option 2 being the preferred alternative because of its greater net benefit.

However, in addition to performing these quantitative cost-benefit analyses, the staff considered several qualitative factors in its regulatory analysis. For each qualitative factor, the staff assigned a qualitative rating to each alternative. This qualitative rating used the number of up-arrows to indicate the impact of considering that qualitative factor on the relative desirability of the alternative. Table H-18 shows these qualitative ratings.

Table H-18 Ratings Assigned to Each Alternative by Qualitative Factor

Qualitative Factor	Option 1	Option 2	Option 3	Option 4
Defense-in-depth		↑	↑↑↑	↑↑
Uncertainties		↑	↑↑↑	↑↑
Severe accident management		↑	↑↑	↑
Hydrogen control		↑↑	↑↑	↑
External events		↑	↑↑	↑↑
Multiunit events		↑	↑↑	↑↑
Independence of barriers		↑	↑↑↑	↑↑
Emergency planning		↑	↑↑↑	↑↑
Consistency between reactor technologies	↑↑↑			↑
Severe accident policy statement	↑↑			↑
International practices		↑	↑↑↑	↑↑

Source: Summarized from SECY-12-0157, Enclosure 1

Note: The analyst should refer to the Commission's response and direction on qualitative factors in SRM-SECY-12-0157 and Appendix A, "Qualitative Factors Assessment Tools," to this NUREG before presenting qualitative factors in this manner.

Summary and Conclusion

The staff determined that many of the qualitative factors supported the following:

- Pursuing an improved venting system for BWRs with Mark I and Mark II containments to address specific design concerns (e.g., high conditional containment failure probability given core melt)

- Providing additional support for severe accident management functions by preventing radiological releases, hydrogen, and steam from entering the reactor building or other locations on the site

- Minimizing the contamination of the site environment

- Reducing the reliance on emergency planning for the protection of public health and safety

Considering both the quantitative cost-benefit analysis results and the qualitative factors, the staff further determined that Options 2 and 3, and most likely Option 4, were cost-justified, based on the substantial increase in overall protection of public health and safety that would be provided by addressing severe accident conditions for BWRs with Mark I and Mark II containments.

Based on its regulatory analysis, the staff concluded that Option 3 (installation of engineered filtered venting systems for Mark I and Mark II containments) was the alternative that would provide the most regulatory certainty and the most timely implementation.

Commission's Response to the Staff's Analysis and Recommendations

The Commission approved Option 2 and directed the staff to further evaluate Options 3 and 4. Enclosure H-4 to this appendix summarizes the staff's further evaluation of Options 3 and 4. The Commission also directed the staff to seek detailed Commission guidance on the use of qualitative factors in a future notation vote paper. In response, the staff submitted SECY-14-0087, "Qualitative Consideration of Factors in the Development of Regulatory Analyses and Backfit Analyses," dated August 14, 2014, and developed Appendix A to this NUREG.

ENCLOSURE H-4: SUMMARY OF DETAILED ANALYSES FOR SECY-15-0085, "EVALUATION OF THE CONTAINMENT PROTECTION AND RELEASE REDUCTION FOR MARK I AND MARK II BOILING-WATER REACTORS RULEMAKING ACTIVITIES"

This enclosure summarizes the detailed analyses supporting the evaluation of containment protection and release reduction strategies for boiling-water reactor (BWR) plants with Mark I and Mark II containments, as documented in SECY-15-0085, "Evaluation of the Containment Protection and Release Reduction for Mark I and Mark II Boiling-Water Reactors Rulemaking Activities," dated June 18, 2015, as well as in NUREG-2206, "Technical Basis for the Containment Protection and Release Reduction Rulemaking for Boiling-Water Reactors with Mark I and Mark II Containments," issued March 2018. The contents of this enclosure should be considered with the previous detailed analyses supporting SECY-12-0157, "Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments," dated November 26, 2012. Enclosure H-3, "Summary of Detailed Analyses for SECY-12-0157, 'Consideration of Additional Requirements for Containment Venting Systems for Boiling Water Reactors with Mark I and Mark II Containments,'" to this appendix summarizes the detailed analyses for SECY-12-0157.

Problem Statement and Regulatory Objectives

The accident that occurred on March 11, 2011, at the Fukushima Dai-ichi nuclear power plant in Japan underscored the importance of reliable operation of containment vents for BWR plants with Mark I and Mark II containments. As part of its response to the lessons learned from this accident, the staff of the U.S. Nuclear Regulatory Commission (NRC) issued Order EA-12-050, "Issuance of Order to Modify Licenses with Regard to Reliable Hardened Containment Vents," dated March 12, 2012. This Order required licensees that operate BWRs with Mark I and Mark II containment designs to install hardened containment vents. These vents would address problems encountered during the Fukushima accident by providing plant operators with improved methods for venting containment during accident conditions and thereby preventing containment overpressurization and subsequent failure. In SECY-11-0137, "Prioritization of Recommended Actions to be Taken in Response to Fukushima Lessons Learned," dated October 3, 2011, the staff also identified an issue involving containment vent filtration and included a recommendation for the addition of an engineered filtered vent system to improve reliability and limit the release of radiological materials if the venting systems are used in a severe accident after the occurrence of significant core damage.

In SECY-12-0157, the staff analyzed whether additional requirements might be warranted to address venting from BWRs with Mark I and Mark II containments after core damage and whether filtering of radiological materials that may be released from the vents would be necessary. The staff evaluated four regulatory options, including (1) the status quo—which served as the regulatory baseline and assumed the staff would continue to implement Order EA-12-050 and install reliable hardened vents to reduce the probability of failure of BWR Mark I and Mark II containments but would take no additional action, (2) upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions, (3) design and install an engineered filtered containment venting system intended to prevent the release of significant amounts of radioactive material following the dominant severe accident sequences at BWRs with Mark I and Mark II containments, and (4) pursue development of requirements and technical acceptance criteria for performance-based severe accident confinement strategies.

The NRC staff provided an evaluation that considered both results from quantitative cost-benefit analyses and qualitative factors related to the four options and recommended that the Commission approve Option 3 to require the installation of an engineered filtering system. While acknowledging that the quantitative analyses indicated the costs of the proposed actions outweighed the benefits, the staff recommended in SECY-12-0157 that the Commission consider both the quantitative and qualitative factors and concluded the proposed additional regulatory actions associated with Option 3 were cost-justified.

In its staff requirements memorandum (SRM) for SECY-12-0157, dated March 19, 2013, the Commission directed the staff to (1) issue a modification to Order EA-12-050 to require BWR licensees with Mark I and Mark II containments to upgrade or replace the reliable hardened vents required by Order EA-12-050 with a containment venting system designed and installed to remain functional during severe accident conditions, and (2) develop technical bases and pursue rulemaking for filtering strategies with drywell filtration and severe accident management of BWR Mark I and Mark II containments. The Commission further ordered that the technical bases should (1) assume that severe-accident-capable vents had been ordered and, as a consequence of that action, should assume that the benefits of these vents accrue equally to engineered filters and to filtration strategies, (2) explore requirements associated with measures to enhance the capability to maintain confinement integrity and to cool core debris, and (3) evaluate multiple performance criteria, including a required decontamination factor and equipment and procedure availability like those required to implement Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54 (hh).[27]

In response to SRM-SECY-12-0157, the staff issued Order EA-13-109, "Issuance of Order To Modify Licenses with Regard to Reliable Hardened Containment Vents Capable of Operation Under Severe Accident Conditions," dated June 6, 2013, which rescinded certain requirements imposed in Order EA-12-050 and required BWR licensees with Mark I and Mark II containments to upgrade or replace their vents with a containment venting system designed and installed to remain functional during severe accident conditions. Order EA-13-109 had two primary requirements that would be implemented sequentially in two phases:

1. Phase 1: Upgrade the venting capabilities from the containment wetwell to provide reliable, severe-accident-capable hardened vents to assist in preventing core damage and, if necessary, to provide venting capability during severe accident conditions.

2. Phase 2: Either install a reliable severe-accident-capable drywell venting system or develop and implement a reliable containment venting strategy that makes it unlikely that a licensee would need to vent from the containment drywell during severe accident conditions.

In response to Order EA-13-109, the severe accident water addition (SAWA) approach required licensees to use water addition in combination with one of two strategies—(1) a severe-accident-capable drywell vent designed to lower temperature limits, or (2) severe accident water management (SAWM) to control the water levels in the suppression pool such that it would be unlikely that a licensee would need to vent from the containment drywell during severe accident conditions (Nuclear Energy Institute (NEI), 2014).

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[27] The SRM for SECY-12-0157 provided additional directions which are addressed in SECY-15-0085.

With the issuance of Order EA-13-109, the staff also began developing the regulatory basis for the containment protection and release reduction (CPRR)[28] rulemaking for BWRs with Mark I and Mark II containments. The objective of the CPRR regulatory basis was to determine what, if any, additional requirements were warranted on filtering strategies and severe accident management for BWRs with Mark I and Mark II containments, assuming the installation of severe-accident-capable hardened vents per Order EA-13-109.

Regulatory Alternatives

The staff interacted with industry and members of the public and identified four major regulatory alternatives comprising numerous subalternatives for choices on filtering strategies and severe accident management for BWRs with Mark I and Mark II containment designs. The four main CPRR regulatory alternatives considered in the regulatory analysis performed in support of SECY-15-0085 were the following:

- Alternative 1: Severe-Accident-Capable Vents (Status Quo). Continue with the implementation of Order EA-13-109 and installation of severe-accident-capable vents, without taking additional regulatory actions related to BWR Mark I and Mark II containments. This alternative represented the status quo and served as the regulatory baseline against which the benefits and costs of other alternatives were measured.

- Alternative 2: Rulemaking to Make Order EA-13-109 Generically Applicable. Pursue rulemaking to make Order EA-13-109 generically applicable to protect BWR Mark I and Mark II containments against overpressurization. The potential benefits associated with this option resulted from making generically applicable the requirements in Order EA-13-109 related to improved reporting, change control, and other aspects of controlling licensing basis information.

- Alternative 3: Rulemaking to Make Order EA-13-109 Generically Applicable and Additional Requirements for SAWA to Address Uncontrolled Releases from Major Containment Failure Modes. Pursue rulemaking to address overall BWR Mark I and Mark II containment protection against multiple failure modes by making Order EA-13-109 generically applicable and requiring external water addition points that would allow water to be added into the reactor pressure vessel (RPV) or drywell to prevent containment failure from both overpressurization and liner melt-through.

- Alternative 4: Rulemaking to Reduce Releases during Controlled Venting (Filtering Strategies, Engineered Filters). Pursue rulemaking to address both containment protection against multiple failure modes and release reduction measures for controlling releases through the containment venting systems. This alternative would make Order EA-13-109 generically applicable and require external water addition into the RPV or drywell. In addition, licensees would be required to reduce the fission products released from containment by (1) implementing strategies to maximize the availability and efficiency of the wetwell in scrubbing or filtering fission products before venting from containment or (2) installing an engineered filter in the containment vent paths (or both).

[28] As the rulemaking progressed, the staff determined that the original rulemaking name (filtering strategies) no longer matched the purpose of the activity. The staff believed it was more logical to have the rulemaking reflect the two issues being analyzed—enhanced containment protection and release reduction.

A CPRR strategy is an action taken before or during a severe accident to protect the containment's structural integrity or to reduce the amount of radiological material released to the environment. Examples include containment venting following core damage (a containment protection strategy) and the installation of engineered filters on the containment vent lines (a release reduction strategy). Such high-level strategies can be divided into more specific categories according to how they are implemented. From the four main regulatory alternatives defined above, 20 regulatory subalternatives were defined by specific combinations of CPRR strategies. These combinations of CPRR strategies considered many factors, including the following:

- Wetwell and drywell venting priority (before and after core damage)

- Venting actuation (before and after core damage)

- Venting operation mode (before and after core damage)

- Vent reclosure if core damage is imminent

- Postaccident water injection location and operating mode

- Filter size and decontamination factor

Table 19 summarizes the 20 regulatory subalternatives, how each subalternative maps to the options defined in SECY-12-0157 and the alternatives defined in SECY-15-0085, and the combinations of CPRR strategies used to distinguish among them.

Safety Goal Evaluation

A safety goal evaluation for Alternative 3 and Alternative 4 was performed in this regulatory analysis because these two main regulatory alternatives were considered generic safety enhancement backfits subject to the substantial additional protection standard at 10 CFR 50.109(a)(3). Each alternative, if implemented, would improve containment performance by reducing (1) the probability of containment failure, given the assumed occurrence of a severe accident scenario, and/or (2) the amount of radiological material released to the environment from a severe accident scenario. However, since none of the alternatives would impact the frequency of core damage accidents (i.e., the change in core damage frequency (CDF) for each alternative relative to the regulatory baseline was zero), the safety goal screening criteria in the regulatory analysis guidelines could not be used to determine whether each alternative could result in a substantial increase in overall protection of public health and safety.

To perform the safety goal evaluation, the staff analyzed numerous regulatory alternatives to directly compare their potential safety benefits to the quantitative health objectives (QHOs) for average individual early fatality risk and average individual latent cancer fatality risk described in the Commission's Safety Goal Policy Statement (NRC, 1986). Each of the alternatives was compared to Alternative 1 (status quo and regulatory baseline) to determine the relative benefits and costs of the alternative.

The staff determined there was zero average individual early fatality risk, conditioned on the assumed occurrence of the modeled severe accident scenarios. In part this resulted from the fact that the modeled accident progression resulted in releases that begin late when compared to the time needed to evacuate members of the public living near the modeled nuclear power plant site.

Table H-19 Summary of Regulatory Subalternatives and Distinguishing Attributes

				Before Core Damage				After Core Damage					
Index	Regulatory Subalternative	SECY-12-0157 Option	SECY-15-0085 Alternative	Venting Priority	Venting Actuation	Venting Operation Mode	Reclose Valve if Core Damage is Imminent	Postaccident Water Injection Location	Postaccident Water Injection Operating Mode	Venting Priority	Venting Actuation	Venting Operation Mode	Filter Size and DF
1	1	2	NA	WWF	M	AV	Yes	NA	NA	WWF	M	OLO	NA
2	2A	2	NA	WWF	M	AV	Yes	NA	NA	WWF	M	OLO	NA
3	3A	2	1,2,3	WWF	M	AV	Yes	RPV	SAWA	WWF	M	OLO	NA
4	3B	2	1,2,3	WWF	M	AV	Yes	DW	SAWA	WWF	M	OLO	NA
5	4Ai(1)	4	4	WWF	M	AV	Yes	RPV	SAWA	WWF	M	VC	NA
6	4Ai(2)	4	4	WWF	M	AV	Yes	DW	SAWA	WWF	M	VC	NA
7	4Aii(1)	4	4	WWF	M	AV	Yes	RPV	SAWM	WWF	M	OLO	NA
8	4Aii(2)	4	4	WWF	M	AV	Yes	DW	SAWM	WWF	M	OLO	NA
9	4Aiii(1)	4	4	WWF	M	AV	Yes	RPV	SAWM	WWF	M	VC	NA
10	4Aiii(2)	4	4	WWF	M	AV	Yes	DW	SAWM	WWF	M	VC	NA
11	4Bi(1)	3	4	WWF	M	AV	Yes	RPV	SAWA	WWF	M	OLO	S
12	4Bi(2)	3	4	WWF	M	AV	Yes	DW	SAWA	WWF	M	OLO	S
13	4Bii	3	4	WWF	M	AV	Yes	DW	SAWA	DWF	M	OLO	S
14	4Biii	3	4	WWF	M	AV	Yes	DW	SAWA	DWF	P	OLO	S
15	4Biv	3	4	DWF	P	OLO	No	DW	SAWA	DWF	P	OLO	S
16	4Ci(1)	3	4	WWF	M	AV	Yes	RPV	SAWA	WWF	M	OLO	L
17	4Ci(2)	3	4	WWF	M	AV	Yes	DW	SAWA	WWF	M	OLO	L
18	4Cii	3	4	WWF	M	AV	Yes	DW	SAWA	DWF	M	OLO	L
19	4Ciii	3	4	WWF	M	AV	Yes	DW	SAWA	DWF	P	OLO	L
20	4Civ	3	4	DWF	P	OLO	No	DW	SAWA	DWF	P	OLO	L

Venting Priority
DWF: drywell first strategy
WWF: wetwell first strategy
Venting Actuation
M: manual
P: passive (rupture disc)
Venting Operation Mode
AV: anticipatory venting
OLO: open at 15 psig and leave open
VC: venting cycling at primary containment pressure limit with 10 psi band

Postaccident Water Injection Location
DW: drywell via external connection
RPV: reactor pressure vessel via external connection
Postaccident Water Injection Operating Mode
SAWA severe accident water addition
SAWM severe accident water management
Filter Size and Decontamination Factor (DF)
L: large with DF of 1000
S: small with DF of 10

(Source: NUREG-2206, Table 2-2)

The staff then performed a screening analysis for the average individual latent cancer fatality risk QHO by evaluating all United States (U.S.) BWRs with Mark I containments (a total of 22 units at 15 sites) and Mark II containments (a total of eight units at five sites). For this screening analysis, the staff developed a conservative high estimate of frequency-weighted average individual latent cancer fatality risk within 10 miles using the following parameter values:

- An extended loss of alternating current power (ELAP)[29] frequency value of $7x10^{-5}$ per reactor-year—which represented the highest value among all BWRs with Mark I and Mark II containments

- A success probability for flexible coping strategies (FLEX) equipment of 0.6 per demand—which assumed implementation of FLEX will successfully mitigate an accident involving an ELAP 6 out of 10 times

- A conditional average individual latent cancer fatality risk of $2x10^{-3}$ per event—which represented the highest value among all BWRs with Mark I and Mark II containments

These assumed parameter values resulted in a conservative high estimate of frequency-weighted individual latent cancer fatality risk within 10 miles of approximately $7x10^{-8}$ per reactor-year, which is greater than an order of magnitude less than the QHO for an average individual latent cancer fatality risk of approximately $2x10^{-6}$ per reactor-year. This conservative high estimate did not take credit for any of the accident strategies and capabilities described in the 20 CPRR alternatives and subalternatives. Figure H-19 shows the incremental benefit for each alternative and subalternative, compared to the status quo and Order EA-13-109. If licensees were to choose to implement SAWA/SAWM as part of compliance with EA-13-109, the uncertainty band for Alternative 3 would apply. However, since EA-13-109 did not specifically require SAWA/SAWM, it was not credited in Figure H-18 for Alternative 1 or Alternative 2.

If an ELAP occurs and results in core damage, an engineered filtered containment venting system would reduce offsite consequences. However, because the average individual latent cancer fatality risk within 10 miles for the status quo alternative (Alternative 1) was already well below the associated QHO, the staff concluded that the design and installation of an engineered filtered containment venting system or a performance-based confinement strategy for BWRs with Mark I and Mark II containments would not meet the threshold for a substantial safety enhancement. Moreover, although this analysis did not include all accident scenarios that a full-scope Level 3 PRA would need to consider, the staff concluded that none of the alternatives could result in a substantial increase in overall protection of public health and safety. Therefore, the staff recommended that rulemaking not be pursued for SECY-12-0157 Option 3 or Option 4. Furthermore, the staff concluded that a detailed regulatory analysis of the various alternatives was not warranted and would provide little additional insight into the regulatory decision because the margin to the QHOs did not support a substantial safety benefit.

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29 An ELAP is defined as a station blackout (SBO) that lasts longer than the SBO coping duration specified in 10 CFR 50.63, "Loss of all alternating current power."

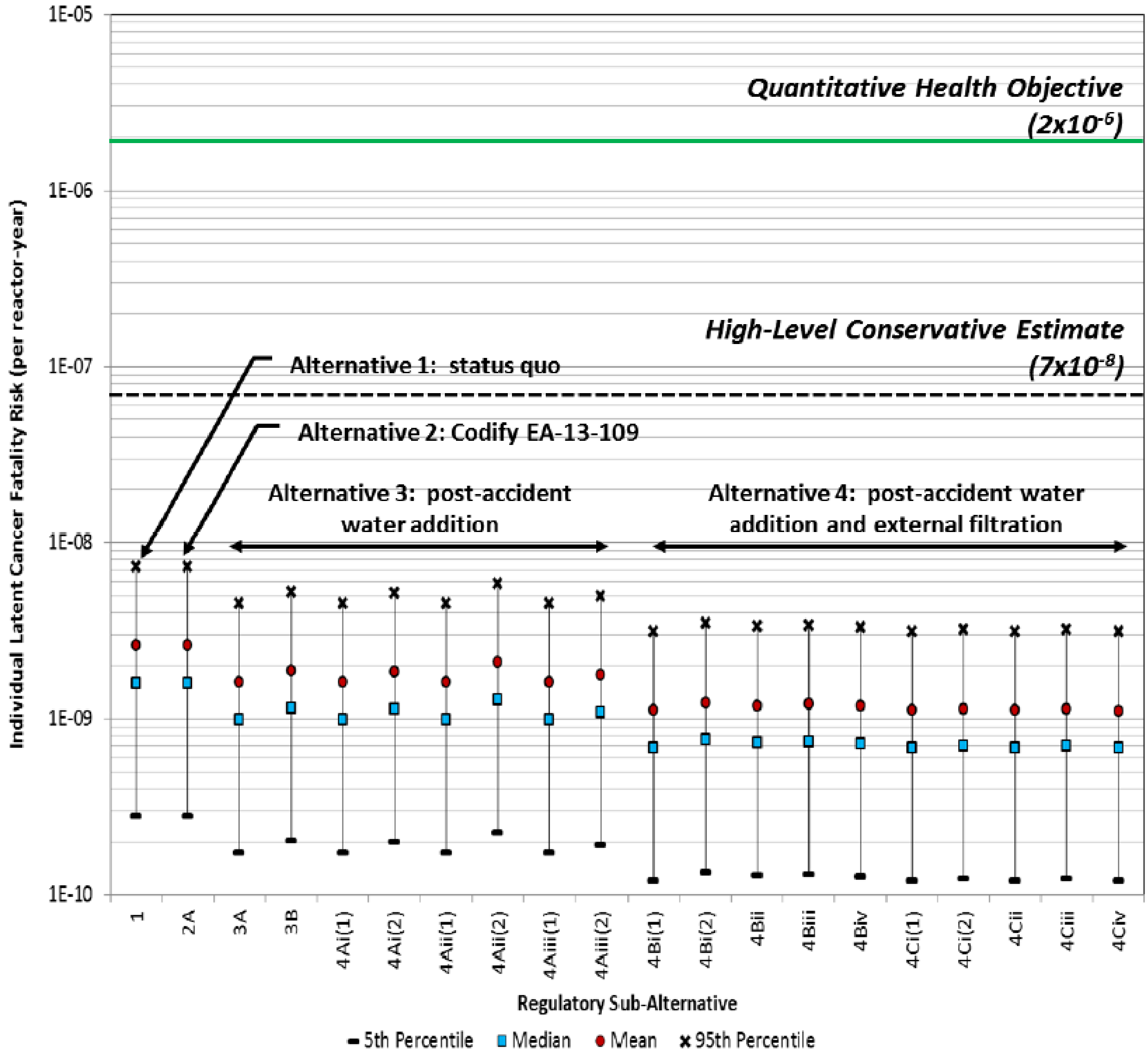


Figure H-18 Uncertainty in Average Individual Latent Cancer Fatality Risk (0–10 miles)
(Source: SECY-15-0085, Enclosure, Figure 3-3)

Technical Evaluation

Accident Scenario Selection

The staff considered the following factors during the development of the technical approach for the accident sequence analysis performed for SECY-15-0085:

- The risk evaluation should provide risk metrics for each of the 20 CPRR regulatory analysis subalternatives, according to the schedule established by the Commission and the resources allotted by NRC management.

- Consistent with the NRC's regulatory analysis guidelines, the risk evaluation should provide fleet-average risk estimates. Therefore, the technical approach should consider the impacts of plant-to-plant variability.

- Consistent with Recommendation 5.1 in the Fukushima Near-Term Task Force (NTTF) report, the accident sequence analysis should focus on accidents initiated by ELAP events.

- The generic estimates of release sequence frequencies and conditional consequences in NUREG/BR-0184, “Regulatory Analysis Technical Evaluation Handbook,” issued January 1997, were developed from previous probabilistic risk assessments (PRAs) that did not consider CPRR strategies and therefore cannot be used to provide an adequate technical basis for the CPRR risk evaluation.

- Core damage event trees (CDETs) should be developed to (1) model the impact of equipment failures and operator actions occurring before core damage that affect severe accident progression and the probability that CPRR strategies are successfully implemented, (2) match the initial and boundary conditions used in the thermal-hydraulic simulation of severe accidents in MELCOR, and (3) probabilistically consider mitigating strategies for beyond-design-basis external events required by Order EA-12-049, “Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events,” dated March 12, 2012.

- The CPRR strategies addressed in the set of 20 regulatory analysis subalternatives are specified at a conceptual level. Therefore, it is acceptable to develop high-level generic accident progression event trees (APETs) to model the CPRR strategies because no information is available about their specific design details.

Analysts used a modular approach to develop the CDETs and APETs, as shown in Figure H-19. This modeling approach streamlined the development of risk estimates.

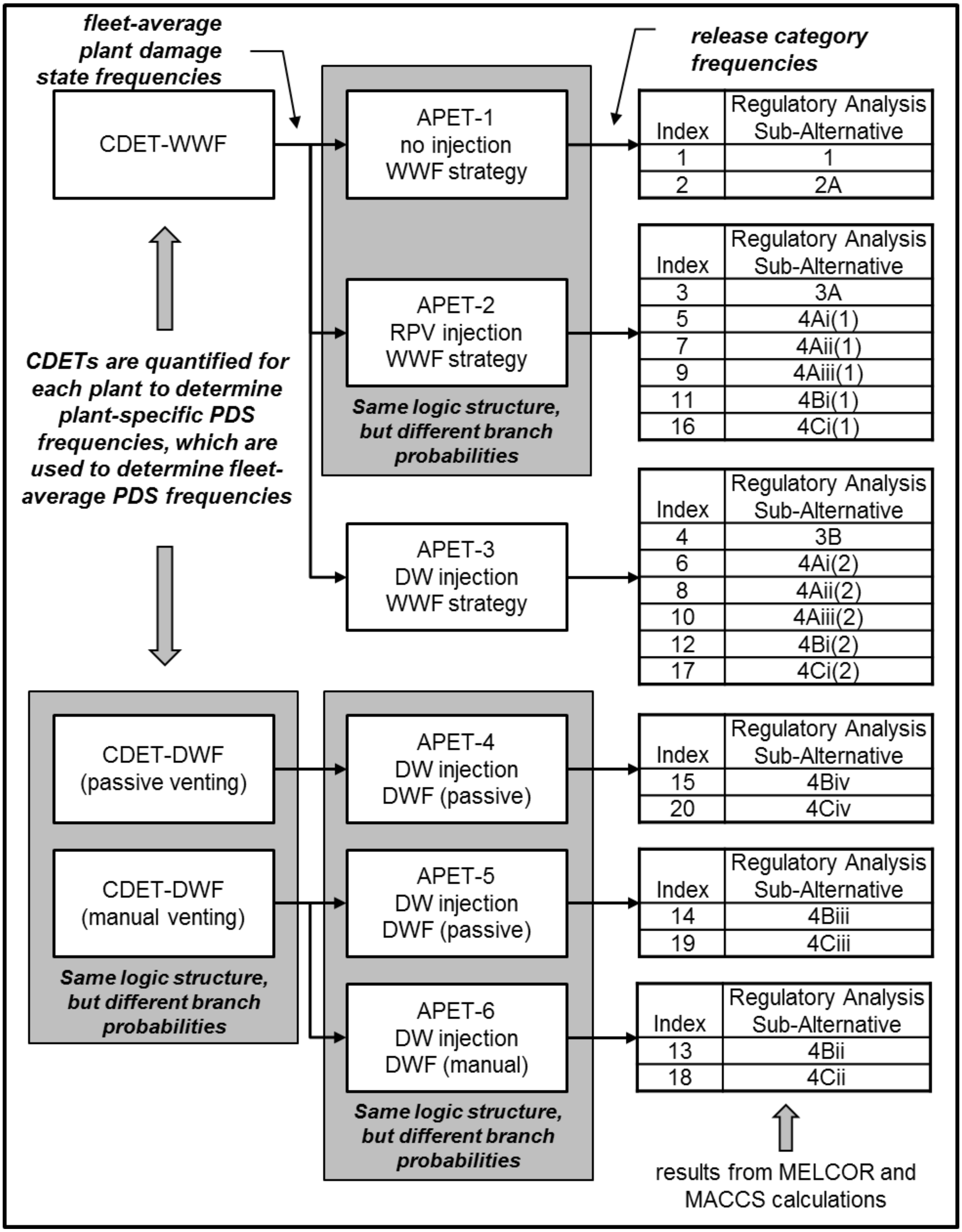


Figure H-19 Modular Approach to Event Tree Development
(Source: NUREG-2206, Figure 2-1)

MELCOR Severe Accident Progression and Source Term Analyses

The MELCOR analyses addressed two main categories: (1) reactor systems and containment thermal-hydraulics under severe accident conditions and (2) assessment of source terms—the timing, magnitude, and other characteristics of fission product releases to the environment. The first category provided insight into the state of containment vulnerability under severe accident conditions and information needed to assess containment integrity. The second category provided information needed to assess the offsite radiological consequences associated with releases of radioactive materials to the environment.

The NRC based the development of the MELCOR calculation matrices (see Table 3-2 and Table 3-3, NRC, 2018b) on the CPRR alternatives defined by the accident sequence analysis. The MELCOR analyses investigated detailed accident progression, containment response, and source terms for representative Mark I and Mark II containment designs following an ELAP. The selection of accident scenarios considered for MELCOR analyses was informed by the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project (see Enclosure H-2, "Summary of the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," to this appendix), the Fukushima Dai-ichi nuclear power plant accident reconstruction study (Sandia National Laboratories, 2012), and the detailed analyses in SECY-12-0157. The representative Mark I containment selected was similar in configuration to Peach Bottom Atomic Power Station (Peach Bottom), Unit 2, and the representative Mark II containment was similar in configuration to LaSalle County Station (LaSalle). The Mark I MELCOR calculation matrix included sensitivity cases to evaluate the impact on results of using plausible alternative assumptions about multiple factors, including (1) mode of venting, (2) status of RPV depressurization, (3) mode of FLEX water injection, and (4) water management. The Mark II MELCOR calculation matrix included a subset of the Mark I matrix, based on the insights from the Mark I MELCOR calculations, and included sensitivity cases to evaluate the impact of the pedestal and lower cavity designs among the fleet by modifying the base model.

The scope and technical approach for the MELCOR analyses performed in support of SECY-15-0085 were similar to those of SECY-12-0157. In both cases, the technical approach considered best estimate modeling of accident progression and incorporated both preventive and mitigative accident management measures, including (1) venting, (2) water addition, water management, or both, and (3) installation of engineered filters. However, an important distinction between the technical approaches is that, in SECY-12-0157, water addition was considered in a generic way because the industry's post-Fukushima Dai-ichi severe accident management strategies were still evolving and the concepts of SAWA and SAWM had not yet emerged. Moreover, the industry was formulating its FLEX strategy for severe accident mitigation applications at the time. By contrast, these various concepts and severe accident management measures were more mature by the time detailed analyses were performed for SECY-15-0085 and were, therefore, considered in developing the technical approach for these analyses.

MACCS Consequence Analyses

Like the MELCOR analyses, the scope and technical approach for the MACCS analyses performed in support of SECY-15-0085 were similar to those of SECY-12-0157. The NRC used MACCS to calculate offsite radiological consequences with site-specific population, economic, land use, weather, and evacuation data for reference Mark I and Mark II sites. The agency selected Peach Bottom and the Limerick Generating Station (Limerick) as the site-specific reference models for the offsite consequence analyses to enable greater modeling fidelity for sites with relatively high population densities (Peach Bottom had the second highest population within a 50-mile radius among the 15 Mark I sites and Limerick had the highest population within a 50-mile radius among the five Mark II sites).

The staff performed offsite consequence analyses for the source terms generated by MELCOR corresponding to different CPRR accident management strategies following an ELAP event. It assessed the relative public health risk reduction associated with various containment protection and release reduction measures with respect to various offsite radiological consequence measures, including (1) average individual early fatality risk and average individual latent cancer fatality risk, (2) population dose, (3) land contamination, (4) economic costs, and (5) displaced population. Land contamination areas and displaced populations represented additional consequence metrics that the staff reported for consideration by decisionmakers, although they are not required as inputs to safety goal evaluations or regulatory analyses. The calculated offsite radiological consequences were weighted by accident frequency to assess relative public health risk reduction.

Tables H-20 and H-21 show the summary MACCS results respectively for the 18 Mark I and the 9 Mark II source term bins. As shown on the tables, the staff reported some consequence metrics out to a 100-mile radius from the plant.

Table H-20 MACCS Results for 18 Mark I Source Term Bins

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individiual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	0	4.65E-07	4.57E-08	2.06E-08	1,620	2,380
2	48DF100	0.002%	0.02%	11.4	8	0	1.90E-06	1.90E-07	8.69E-08	5,480	8,260
3	10DF100	0.01%	0.08%	16.3	6	0	6.25E-06	7.16E-07	3.21E-07	16,500	27,300
4	7DF1000	0.02%	0.26%	14.9	20	0	1.72E-05	2.35E-06	1.01E-06	48,400	77,600
5	11DF10	0.06%	0.78%	14.4	4	0	2.03E-05	3.36E-06	1.62E-06	71,200	127,000
6	48	0.23%	1.69%	11.4	8	0	7.95E-05	1.61E-05	7.79E-06	253,000	450,000
7	15	0.60%	5.85%	14.9	7	0	1.21E-04	3.28E-05	1.64E-05	524,000	932,000
8	46	0.98%	11.01%	14.8	17	0	1.53E-04	4.59E-05	2.34E-05	790,000	1,410,000
9	5DF10	1.05%	2.89%	24.2	34	0	3.55E-04	7.50E-05	3.35E-05	1,040,000	1,720,000
10	5	1.39%	6.46%	24.2	41	0	4.06E-04	9.78E-05	4.51E-05	1,360,000	2,290,000
11	8	1.49%	19.25%	14.9	5	0	1.35E-04	6.41E-05	3.43E-05	1,110,000	2,030,000
12	1	1.93%	22.68%	14.9	22	0	2.91E-04	1.01E-04	5.23E-05	1,720,000	3,090,000
13	41DF1000	3.40%	7.65%	9.8	17	0	5.22E-04	1.49E-04	7.89E-05	1,900,000	3,610,000
14	22dw	2.82%	18.64%	14.9	27	0	4.27E-04	1.28E-04	6.57E-05	1,830,000	3,320,000
15	53	2.79%	29.05%	17.4	13	0	2.59E-04	1.19E-04	6.96E-05	1,740,000	3,520,000
16	41	4.54%	14.10%	9.8	16	0	5.57E-04	1.75E-04	9.82E-05	2,300,000	4,520,000
17	3DF10	8.85%	24.65%	9.8	63	0	7.10E-04	2.95E-04	1.68E-04	3,830,000	7,720,000
18	52	15.90%	34.32%	17.4	11	0	5.39E-04	2.23E-04	1.50E-04	3,080,000	6,870,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	28DF1000	0.0006%	0.006%	14.9	7	78,900,000	78,900,000	0	0	-	-
2	48DF100	0.002%	0.02%	11.4	8	79,700,000	79,700,000	1	1	0	0
3	10DF100	0.01%	0.08%	16.3	6	98,100,000	98,700,000	10	11	1	1
4	7DF1000	0.02%	0.26%	14.9	20	141,000,000	141,000,000	23	23	7	7
5	11DF10	0.06%	0.78%	14.4	4	220,000,000	240,000,000	41	65	118	118
6	48	0.23%	1.69%	11.4	8	1,150,000,000	1,390,000,000	116	175	3,440	3,440
7	15	0.60%	5.85%	14.9	7	2,740,000,000	3,690,000,000	190	361	15,000	16,600
8	46	0.98%	11.01%	14.8	17	3,760,000,000	5,220,000,000	242	506	20,700	27,400
9	5DF10	1.05%	2.89%	24.2	34	7,290,000,000	8,600,000,000	351	429	35,200	35,200
10	5	1.39%	6.46%	24.2	41	9,900,000,000	12,000,000,000	479	715	51,400	51,500
11	8	1.49%	19.25%	14.9	5	5,960,000,000	9,720,000,000	286	673	40,500	55,800
12	1	1.93%	22.68%	14.9	22	13,000,000,000	17,400,000,000	549	1,040	64,500	79,700
13	41DF1000	3.40%	7.65%	9.8	17	19,400,000,000	24,700,000,000	783	1,170	168,000	190,000
14	22dw	2.82%	18.64%	14.9	27	12,900,000,000	18,300,000,000	544	1,010	93,700	114,000
15	53	2.79%	29.05%	17.4	13	15,700,000,000	26,500,000,000	573	1,290	111,000	142,000
16	41	4.54%	14.10%	9.8	16	25,500,000,000	35,400,000,000	904	1,500	235,000	281,000
17	3DF10	8.85%	24.65%	9.8	63	47,000,000,000	68,100,000,000	1,360	2,470	417,000	504,000
18	52	15.90%	34.32%	17.4	11	46,500,000,000	87,700,000,000	987	2,170	467,000	873,000

* Note: To quantify the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5 percent of that source term's total cumulative cesium release to the environment. Cesium, rather than iodine, was selected here because all of the resulting offsite consequences are driven by long-term phase exposures.

(Source: NUREG-2206, Table 4-22)

Table H-21 MACCS Results for 9 Mark II Source Term Bins

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)	
						0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	0	9.72E-08	1.03E-08	3.45E-09	282	345
2	5DF1000	0.0006%	0.005%	32.2	20	0	1.15E-06	1.81E-07	6.35E-08	4,340	5,440
3	42DF100	0.0043%	0.037%	14.3	13	0	6.58E-06	8.67E-07	3.02E-07	20,700	26,700
4	11	0.042%	0.45%	20.3	20	0	7.90E-05	9.68E-06	3.27E-06	202,000	261,000
5	51DF10	0.23%	2.01%	16.6	9	0	1.35E-04	3.39E-05	1.21E-05	689,000	888,000
6	5	0.55%	4.94%	32.2	20	0	2.29E-04	1.05E-04	4.01E-05	2,160,000	2,900,000
7	3	1.09%	10.26%	14.3	20	0	3.08E-04	1.88E-04	7.43E-05	4,140,000	5,580,000
8	1	2.46%	19.81%	22.8	25	0	4.70E-04	3.17E-04	1.25E-04	6,110,000	8,260,000
9	52	3.57%	28.67%	16.6	10	0	4.03E-04	2.46E-04	1.01E-04	5,430,000	7,440,000

Bin	Rep Case	Rep Case Cs (%)	Rep Case I (%)	Start Time (hrs)	# Hrs with Significant Cs Release*	Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
						0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	11DF1000	0.00004%	0.0005%	20.3	20	381,000,000	381,000,000	-	-	-	-
2	5DF1000	0.0006%	0.005%	32.2	20	381,000,000	381,000,000	0	0	-	-
3	42DF100	0.0043%	0.037%	14.3	13	393,000,000	393,000,000	2	2	0	0
4	11	0.042%	0.45%	20.3	20	844,000,000	846,000,000	44	47	1,030	1,030
5	51DF10	0.23%	2.01%	16.6	9	4,250,000,000	4,380,000,000	130	221	15,400	15,400
6	5	0.55%	4.94%	32.2	20	24,000,000,000	28,000,000,000	303	551	62,400	62,400
7	3	1.09%	10.26%	14.3	20	80,800,000,000	105,400,000,000	698	1,200	619,000	649,000
8	1	2.46%	19.81%	22.8	25	85,500,000,000	109,300,000,000	854	1,680	721,000	741,000
9	52	3.57%	28.67%	16.6	10	53,600,000,000	63,800,000,000	618	1,400	414,000	449,000

* Note: To quantify the time signature of a source term release, an hourly plume segment is considered "significant" if it contributes at least 0.5 percent of that source term's total cumulative cesium release to the environment. Cesium, rather than iodine, was selected here because all the resulting offsite consequences are driven by long-term phase exposures.

(Source: NUREG-2206, Table 4-23)

The offsite radiological consequence estimates for SECY-15-0085 were like those of SECY-12-0157. However, an important distinction between the detailed analyses for SECY-15-0085 and SECY-12-0157 is the use of different performance criteria to evaluate the offsite radiological consequence results. Although not explicitly stated, the detailed analyses for SECY-12-0157 implicitly assumed decontamination factor (DF) as a performance criterion. Specifically, consistent with international nuclear safety practices and guidelines, a DF value of 1,000 was established as a performance target. This is equivalent to one-tenth of one percent of cesium release to the environment and serves as an indirect measure of latent cancer fatality risk and land contamination risk. By contrast, SECY-15-0085 defined six performance criteria related to the attributes of (1) conditional containment failure probability, (2) DF, (3) equipment and procedure availability, (4) total population dose, (5) margin to the QHOs, and (6) long-term relocation. Ultimately, the detailed analyses for SECY-15-0085 used the margin to the safety goal QHOs for average individual early fatality risk within 1 mile and average individual latent cancer fatality risk within 10 miles as the performance criteria to determine whether each alternative could result in a substantial increase in the overall protection of public health and safety.

Risk Evaluation

The staff expanded the scope and level of detail of the PRA model developed for SECY-12-0157 for the detailed analyses for SECY-15-0085. The PRA model used in SECY-12-0157 did not delineate core damage accident sequences. Instead, it relied on a generic estimate of CDF developed from previous NRC staff and licensee PRAs. To provide a quantitative basis for regulatory decisionmaking, the PRA performed in support of SECY-15-0085 included the following features:

- Models to estimate the frequency of ELAP events resulting from internal events and earthquakes, based on industry-developed re-evaluations of seismic hazard estimates.

- CDETs that delineate accident sequences from the occurrence of an ELAP event to the onset of core damage. The CDETs reflect SBO mitigation strategies using installed plant and portable equipment.

- APETs that delineate accident sequences from the onset of core damage to the release of radioactive materials to the environment. The APETs reflect CPRR strategies such as post-core-damage containment venting and water addition.

- Models that include random and seismically-induced equipment failures.

- In-control room and local manual operator actions consistent with emergency operating procedures and severe accident management guidelines.

- Models that identify important contributors to CDF.

- Sensitivity analyses to gain insight into how plausible alternative assumptions about human error probability estimates impact the quantitative results.

These revisions to the PRA model resulted in a lower value for conditional CDF, conditioned on the assumed occurrence of an ELAP, than was reported in SECY-12-0157. The model calculated the CDF caused by ELAPs to be 8.9×10^{-6} per reactor-year, which was about two times lower than the value of 1.6×10^{-5} that SECY-12-0157 estimated. The CDF calculation averaged together the CDF for each BWR plant that was included in the scope of the accident sequence analysis.

Table H-22 summarizes the risk estimates of each regulatory analysis subalternative. These risk estimates represent the point estimate, baseline-case results.

Table H-22 Risk Estimates by Regulatory Analysis Subalternative

Index	Regulatory Analysis Sub-Alternative	Fraction of Core-Damage Frequency		Individual Early Fatality Risk (/y)	Individual Latent Cancer Fatality Risk (/y)			Population Dose (person-rem/y)		Offsite Cost ($ 2013/y)		Land Exceeding Long-Term Habitability Criterion (square miles/y)		Population Subject to Long-Term Protective Actions (persons/y)	
		Vented	Uncontrolled Release	0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
1	1	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
2	2A	0%	100%	0.0E+00	3.0E-09	8.6E-10	4.2E-10	1.3E+01	2.3E+01	9.9E+04	1.3E+05	4.4E-03	7.6E-03	5.1E-01	5.8E-01
3	3A	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
4	3B	42%	58%	0.0E+00	2.1E-09	6.7E-10	3.4E-10	1.1E+01	1.9E+01	7.4E+04	1.0E+05	3.4E-03	6.4E-03	4.1E-01	4.9E-01
5	4Ai(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
6	4Ai(2)	42%	58%	0.0E+00	2.1E-09	6.1E-10	3.1E-10	9.5E+00	1.7E+01	6.8E+04	9.0E+04	3.2E-03	5.8E-03	3.6E-01	4.1E-01
7	4Aii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
8	4Aii(2)	42%	58%	0.0E+00	2.4E-09	7.7E-10	3.9E-10	1.2E+01	2.2E+01	8.9E+04	1.2E+05	3.9E-03	7.3E-03	4.8E-01	5.8E-01
9	4Aiii(1)	58%	42%	0.0E+00	1.8E-09	5.5E-10	2.7E-10	8.6E+00	1.5E+01	6.5E+04	8.5E+04	2.9E-03	5.0E-03	3.3E-01	3.9E-01
10	4Aiii(2)	42%	58%	0.0E+00	2.0E-09	5.6E-10	2.7E-10	8.7E+00	1.5E+01	6.2E+04	7.9E+04	3.0E-03	5.1E-03	3.1E-01	3.4E-01
11	4Bi(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
12	4Bi(2)	42%	58%	0.0E+00	1.4E-09	3.3E-10	1.5E-10	4.8E+00	8.2E+00	3.1E+04	3.8E+04	1.8E-03	2.7E-03	1.6E-01	1.6E-01
13	4Bii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.6E+00	7.9E+00	3.0E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
14	4Biii	42%	58%	0.0E+00	1.4E-09	3.2E-10	1.5E-10	4.7E+00	8.1E+00	3.1E+04	3.7E+04	1.7E-03	2.6E-03	1.5E-01	1.6E-01
15	4Biv	40%	60%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.6E+00	7.8E+00	3.0E+04	3.6E+04	1.7E-03	2.6E-03	1.5E-01	1.5E-01
16	4Ci(1)	58%	42%	0.0E+00	1.3E-09	3.1E-10	1.5E-10	4.5E+00	7.8E+00	2.9E+04	3.7E+04	1.6E-03	2.5E-03	1.5E-01	1.6E-01
17	4Ci(2)	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.5E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
18	4Cii	42%	58%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.4E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01
19	4Ciii	42%	58%	0.0E+00	1.3E-09	3.1E-10	1.4E-10	4.4E+00	7.6E+00	3.0E+04	3.7E+04	1.6E-03	2.4E-03	1.5E-01	1.6E-01
20	4Civ	40%	60%	0.0E+00	1.3E-09	3.0E-10	1.4E-10	4.3E+00	7.4E+00	2.9E+04	3.6E+04	1.5E-03	2.3E-03	1.5E-01	1.5E-01

(Source: NUREG-2206, Table 5-1)

In addition to these point estimate baseline-case results, the staff conducted uncertainty and sensitivity analyses. The staff performed a parametric Monte Carlo uncertainty analysis to gain additional perspective into the uncertainty of the point estimate risk evaluation results. The uncertainty analysis considered seismic hazard curves, seismic fragility curves, random equipment failures, operator actions, and consequences. Table H-23 summarizes information used to perform the parametric uncertainty analysis. Figure H-19 shows the results of the uncertainty analysis.

Table H-23 Uncertainty Analysis Inputs

Events	Distribution	Remarks
Frequency of ELAPs due to internal events	Lognormal Mean = point estimate Error factor =15	An error factor of 15 maximizes the ratio of the 95th percentile to the mean value. This approach does not explicitly consider the uncertainty in the offsite power recovery curves or the uncertainty in the EPS reliability parameters (failure rate and failure-on-demand probability).
Seismic hazard curves	Lognormal	Normal parameters were developed for each point on the seismic hazard curve using the fractile information provided by licensees in their responses to the 10 CFR 50.54(f) information request concerning NTTF Recommendation 2.1.
Seismic fragilities	Double lognormal, using the developed values of C_{50}, β_R, and β_U	Traditional approach to modeling uncertainty in seismic fragility.
Hardware-related failures	Lognormal Mean = point estimate Error factor = 15	An error factor of 15 maximizes the ratio of the 95th percentile to the mean value.
Human failure events	Constrained non-informative prior	A constrained non-informative prior distribution is a beta distribution with mean = point estimate and α = 0.5.
Conditional consequences	Lognormal Mean = point estimate Error factor = 10	Informed by preliminary results of the SOARCA uncertainty analysis project.[a]

[a] NUREG/CR-7155 (draft), “State-of-the-Art Reactor Consequence Analyses Project, Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station.”
(Source: NUREG-2206, Table 5-2)

Staff also performed MACCS sensitivity calculations to analyze the influence of site to site variation. The following sensitivities were conducted:

- Population (low, medium, high)
- Evacuation delay (1 hour)
- Nonevacuating cohort size (5 percent of emergency planning zone population)
- Intermediate phase duration (0, 3 months, and 1 year)
- Long-term habitability criterion (500 mrem per year and 2 rem per year), which can vary among states in the U.S.

A final sensitivity calculation examined evacuation delays on the risk to determine the influence of the plume arrival time on the evacuating population (base case, 3 hour delay, 6 hour delay, no evacuation).

The results of these sensitivity analyses appear in a series of tables in Chapter 4 of NUREG-2206, which report the ratio of the consequences for the sensitivity cases compared to the baseline cases. Table H-24 below shows an example of these sensitivity results tables, analyzing the effect of different site files (different populations) on the baseline-case results. The results show that individual latent cancer fatality risk is relatively insensitive to site file data (variations are within 60 percent). Population dose is directly related to population size, so the sensitivity cases show a strong increase in population dose for larger population site files. For example, for the Mark II high source term, the high site file case has a population dose about 11 times higher than the low site file case. For a given source term, the total offsite cost also increases with higher population site files.

Table H-24 Results for Baseline Cases with Different Site Files

Base Model	Source Term	Site File	Individual Early Fatality Risk	Individual Latent Cancer Fatality Risk			Population Dose (person-rem)		Offsite Cost ($ 2013)		Land (sq mi) Exceeding Long-Term Habitability Criterion		Population Subject to Long-Term Protective Actions	
			0-1.3 mi and beyond	0-10 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi	0-50 mi	0-100 mi
Mark I - Peach Bottom	Mark I - Low (Bin 3)	Med (VT Yankee) / Low (Hatch)	Individual early fatality risk is zero for all baseline and sensitivity cases.	1.52	0.98	0.90	0.92	1.19	2.79	2.75	0.39	0.43	6.20	6.20
		High (Peach Bottom) / Low (Hatch)		0.94	0.74	0.96	2.82	2.07	4.65	4.57	1.53	1.45	2.07	2.07
	Mark I - Med (Bin 10)	Med (VT Yankee) / Low (Hatch)		1.25	0.98	0.97	1.88	2.37	3.08	3.60	0.67	0.72	2.91	2.92
		High (Peach Bottom) / Low (Hatch)		1.02	0.83	1.02	5.83	4.00	8.84	8.22	1.28	1.08	7.15	7.15
	Mark I - High (Bin 17)	Med (VT Yankee) / Low (Hatch)		1.23	1.05	1.08	2.26	3.33	3.58	4.95	0.82	0.82	3.11	4.16
		High (Peach Bottom) / Low (Hatch)		1.00	0.89	1.00	6.78	5.04	11.11	9.33	1.11	0.98	9.96	9.59
Mark II - Limerick	Mark II - Low (Bin 2)	Med (Susquehanna) / Low (Columbia)		1.20	0.93	0.49	0.70	1.00	4.90	4.90	3.93	3.93	*	*
		High (Limerick) / Low (Columbia)		1.63	1.10	0.69	2.33	2.25	20.48	20.48	12.79	12.79	*	*
	Mark II - Med (Bin 5)	Med (Susquehanna) / Low (Columbia)		0.94	0.86	0.49	1.38	1.96	2.32	2.33	0.40	0.56	6.35	6.35
		High (Limerick) / Low (Columbia)		1.17	1.03	0.65	6.53	4.82	11.71	10.63	0.52	0.61	28.96	28.96
	Mark II - High (Bin 8)	Med (Susquehanna) / Low (Columbia)		0.89	0.85	0.59	2.06	3.71	3.07	6.60	0.61	0.76	3.00	3.42
		High (Limerick) / Low (Columbia)		1.07	1.04	0.68	10.82	9.32	18.49	17.97	0.69	0.75	17.87	17.09

* Indicates that both the numerator and denominator in the ratio are zero
(Source: NUREG-2206, Table 4-36)

Cost-Benefit Analysis Results

Although the potential benefits from possible measures to limit releases through the containment venting systems during severe accidents were well below the NRC's threshold for developing regulatory requirements, the staff reported updated industry cost estimates for implementing the CPRR alternatives in SECY-15-0085. However, these updated cost estimates did not change the staff's conclusion from SECY-12-0157 that none of the proposed regulatory alternatives would satisfy the substantial additional protection standard at 10 CFR 50.109 (a)(3).

Summary and Conclusion

The staff developed a risk evaluation and evaluated alternative courses of action related to filtering strategies and severe accident management of BWRs with Mark I and Mark II containments relative to the safety goal QHOs. The staff determined that the possible plant modifications (e.g., engineered filters) to enhance containment protection and release reduction capability beyond those imposed by Order EA-13-109 could result in reductions in offsite consequences. However, these reductions would not meet the quantitative threshold for a substantial safety enhancement because the average individual early fatality risk and average individual latent cancer fatality risk are well below the QHOs without additional plant modifications.

Based on the results of the detailed analyses for SECY-15-0085, the staff planned to proceed with Alternative 3: Rulemaking to Make Order EA-13-109 Generically Applicable and Additional Requirements for SAWA to Address Uncontrolled Releases from Major Containment Failure Modes. The rulemaking would include the planned implementation of Phase 2 of the order to require licensees of BWRs with Mark I and Mark II containments to have the capability to add water from external sources and control the flow to cool core debris during severe accident conditions. The staff concluded that the ability to provide post-core-damage water addition results in worthwhile additional protection for public health and safety by: (1) protecting the integrity of the containment; (2) reducing the release of radioactive materials in some severe accident scenarios; and (3) contributing to the balance between accident prevention and mitigation.

The staff's plan to proceed with Alternative 3 for the CPRR rulemaking differed from the staff's recommendation in SECY-12-0157 to require the installation of an engineered filtering system. More detailed analyses resulted in the following findings:

- The CDF from an ELAP event was lower than estimated in SECY-12-0157.

- The identification of important contributors to CDF and sensitivity analyses enhanced the staff's confidence in its quantitative analyses and therefore reduced the importance of remaining uncertainties.

- External water addition was shown to avert containment failure and achieve benefits in terms of averted health risks in a wider range of scenarios than an engineering filtering system (e.g., in scenarios where the release pathway bypasses the filtering system).

Therefore, the staff recommended proceeding with a proposed rulemaking to address the containment protection improvements related to venting and water addition without including requirements for installing engineered filtering systems.

Commission's Response to the Staff's Analysis and Recommendations

The Commission disapproved the staff's plan to proceed with Alternative 3. Instead, the Commission approved Alternative 1, which was to continue with the implementation of Order EA-13-109 and installation of severe-accident-capable vents (including SAWA/SAWM as part of Phase 2 compliance with the Order), without taking additional regulatory actions related to BWR Mark I and Mark II containments. The reasoning for this decision was articulated in the Chairman's comments in the Commission Voting Record. The Chairman noted that "there is no practical difference in safety outcomes between Alternatives 1 and 3…Order EA-13-109, which was imposed on all BWRs with Mark I and II containments in 2013, already serves as a legally binding mechanism that effectively achieves the results the staff is seeking…[Furthermore] there are no expectations that a BWR with a Mark I or II containment will ever be licensed to operate in the United States again," which obviated the need to expend agency resources to make Order EA-13-109 generically applicable through rulemaking (NRC, 2015b).

The Commission further directed the staff to leverage the draft regulatory basis to the extent applicable to support resolution of the post-Fukushima Dai-ichi Tier 3 item related to containments of other designs (NTTF Recommendation 5.2). The NTTF Recommendation 5.2 was subsequently closed by SECY-16-0041, “Closure of Fukushima Tier 3 Recommendations Related to Containment Vents, Hydrogen Control, and Enhanced Instrumentation,” dated March 31, 2016, with no further regulatory action.

ENCLOSURE H-5: SUMMARY OF DETAILED ANALYSES FOR SECY-13-0112 AND NUREG-2161, "CONSEQUENCE STUDY OF A BEYOND-DESIGN-BASIS EARTHQUAKE AFFECTING THE SPENT FUEL POOL FOR A U.S. MARK I BOILING-WATER REACTOR"

This enclosure summarizes the detailed analyses supporting the evaluation of expedited spent fuel transfer from the spent fuel pool (SFP) to dry cask storage for a reference plant, as documented in SECY-13-0112, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," dated October 9, 2013, and in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor." The contents of this enclosure should be considered with the subsequent detailed analyses supporting COMSECY-13-0030, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel." Enclosure H-6, "Summary of Detailed Analyses in COMESECY-13-0030, Enclosure H-1, 'Regulatory Analysis for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel,'" to this appendix summarizes the detailed analyses for COMSECY-13-0030.

Problem Statement and Regulatory Objectives

Previous risk studies have shown that storage of spent fuel in a high-density configuration in SFPs is safe and that the risk is appropriately low (see for example, NUREG-1738, "Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants"). These studies used simplified and sometimes bounding assumptions and models to characterize the likelihood and consequences of beyond-design-basis accidents involving SFPs. As part of the Nuclear Regulatory Commission's (NRC's) post-9/11 security assessments, detailed thermal-hydraulic and severe accident progression models for SFPs were developed and applied to assess the realistic heatup of spent fuel under various pool draining conditions. In 2009, together with these post-9/11 security assessments, the NRC issued additional regulatory requirements codified in Title 10 of the *Code of Federal Regulations* (10 CFR) Part 50, Section 54, "Conditions of licenses." In particular, 10 CFR 50.54(hh)(2) requires that each reactor licensee develop and implement guidance and strategies intended to maintain or restore core cooling, containment, and SFP cooling capabilities under conditions associated with certain beyond-design-basis events.

Following the 2011 accident at the Fukushima Dai-ichi nuclear power plant in Japan that resulted from the Tohoku earthquake and tsunami, several stakeholders submitted comments to the NRC Commission and staff requesting that regulatory action be taken to require the expedited transfer of spent fuel stored in SFPs to dry casks. The basis for these requests was that expediting the transfer of spent fuel in SFPs to dry casks would reduce the potential consequences associated with a loss of SFP coolant inventory by decreasing the amount of spent fuel stored in affected SFPs, thereby decreasing the heat generation rate and radionuclide source term associated with affected spent fuel. In response to Commission direction in staff requirements memorandum (SRM)-SECY-12-0025, "Staff Requirements—SECY-12-0025—Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Tsunami," dated March 9, 2012, the staff implemented regulatory actions that originated from the Near-Term

Task Force (NTTF) recommendations to enhance reactor and SFP safety. The staff issued two orders requiring enhancements to SFP safety:

1. Order EA-12-049, "Issuance of Order to Modify Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012, which requires that licensees develop, implement, and maintain guidance and strategies to maintain or restore core cooling, containment, and SFP cooling capabilities following a beyond-design-basis external event.

2. Order EA-12-051, "Order Modifying Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012, which requires that licensees install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event.

The results are based on previous risk studies without these enhancements, in which the staff had concluded that existing requirements for both SFPs and dry casks provide adequate protection of public health and safety. However, in response to events following the accident at Fukushima, the staff determined that it should (1) confirm that high-density SFP configurations continue to provide adequate protection of public health and safety; and (2) assess potential safety benefits (or detriments) and costs associated with expediting the transfer of spent fuel from the SFP to dry casks at a reference plant with a boiling-water reactor (BWR) and Mark I containment design (the same type of reactor involved in the Fukushima Dai-ichi nuclear power plant accident).

Regulatory Alternatives

The regulatory analyses performed in support of SECY-13-0112 and NUREG-2161 considered the following two regulatory alternatives that address spent fuel storage requirements:

1. Option 1: Maintain Existing Spent Fuel Storage Requirements (Status Quo). This alternative reflected the Commission decision not to expedite the storage of spent fuel from SFPs to dry casks but to continue with the NRC's existing regulatory requirements for spent fuel storage. Under this alternative, spent fuel is moved into dry storage only as necessary to accommodate fuel assemblies being removed from the core during refueling operations. It also assumed that all applicable requirements and guidance to date had been implemented, but no implementation was assumed for related generic issues or other staff requirements or guidance that were unresolved or still under review at the time of the analysis. This alternative assumed (1) continued storage of spent fuel in high-density racks within a relatively full SFP, and (2) compliance with all current regulatory requirements, including those described above for 10 CFR 50.54(hh)(2), Order EA-12-049, and Order EA-12-051.[30] Furthermore, because SFPs have a limited amount of available storage—even after licensees expanded their storage capacity using high-density storage racks—the alternative assumed that the existing practice of transferring spent fuel from SFPs to casks in accordance with 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and

[30] Although Option 1 assumed compliance with the post-Fukushima mitigation strategies required under Order EA-12-049 and the reliable SFP instrumentation required under Order EA-12-051, this was not explicitly modeled as part of the study. Instead, compliance with these requirements was treated as a qualitative factor that would significantly enhance the likelihood of successful mitigation, and thereby reduce the conditional probability of radiological release under Option 1.

High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste," would continue. This alternative represented the status quo and served as the regulatory baseline against which the costs and benefits of Option 2 were measured.

2. Option 2: Expedited Spent Fuel Transfer to Achieve Low-density SFP Storage. This alternative assumed that older spent fuel assemblies would be expeditiously moved from SFP storage to dry cask storage beginning in 2014 to achieve and maintain a low-density loading of spent fuel in existing high-density racks within 5 years. It did not evaluate re-racking of the SFP to a low-density rack configuration because such a situation was judged to be inefficient in terms of regulatory benefit, given that much of the benefit could be achieved by storing less fuel in the existing high-density racks. Because of the low-density SFP loading, this alternative had a smaller long-lived radionuclide inventory in the SFP, a lower overall heat load in the SFP, and a slight increase in the initial water inventory that displaced the removed spent fuel assemblies.

The staff recognized potential cost and risk impacts associated with the transfer of spent fuel from SFPs to dry casks after 5 years of cooling and during long-term dry cask storage. If included, these cost and risk impacts would have reduced the overall net benefit of Option 2 relative to Option 1. However, these effects were conservatively ignored to calculate the potential benefit per reactor-year by comparing only the safety of high-density SFP storage to low-density SFP storage and its implementation costs.

Safety Goal Evaluation

To perform the safety goal evaluation, the staff analyzed the regulatory alternatives to directly compare their potential safety benefits to the quantitative health objectives (QHOs) for average individual early fatality risk and average individual latent cancer fatality risk described in the Commission's Safety Goal Policy Statement (NRC, 1986).

Since the reactor building that houses the SFP does not provide a containment barrier like the containment structure surrounding the reactor core—especially under conditions postulated to dominate the release of radioactive materials from spent fuel—the staff assumed the frequency of a release of radioactive material to the environment would be the same as the frequency of spent fuel damage. Under this assumption, the radiological release frequency was estimated to range from 7×10^{-7} to 5×10^{-6} per reactor-year, when considering all initiators that could challenge SFP cooling or integrity.

Despite the large releases for certain predicted accident progressions, the staff determined there was zero average individual early fatality risk, conditioned on the assumed occurrence of the modeled severe accident scenarios. In part, this was because the modeled accident progressions resulted in releases that begin late relative to the time needed to evacuate members of the public living near the modeled nuclear power plant site.

Using the upper limit of the spent fuel damage and radiological release frequency of 5×10^{-6} per reactor-year combined with a conditional average individual latent cancer fatality risk within 10 miles of 4×10^{-4} resulted in a bounding average individual latent cancer fatality risk of 2×10^{-9} per reactor-year. This calculated value was about 3 orders of magnitude below the QHO of 2×10^{-6} per reactor-year for an average individual latent cancer fatality risk within 10 miles. The staff therefore concluded that Option 2 could not result in a substantial increase in overall protection of public health and safety.

Technical Evaluation

The staff performed detailed analyses using state-of-the-art, validated, deterministic methods and assumptions, supplemented with probabilistic insights where practical.

The study considered two SFP configurations:

1. High-density Loading Configuration: A relatively full SFP in which the hottest spent fuel assemblies are surrounded by four cooler fuel assemblies in a 1x4 loading pattern throughout the pool[31]

2. Low-density Loading Configuration: A minimally loaded pool in which all spent fuel with at least 5 years of pool cooling has been removed to ensure the hottest fuel assemblies are surrounded by additional water

To evaluate the potential benefits of mitigation strategies required in 10 CFR 50.54 (hh)(2), the study analyzed each loading configuration for two different cases—(1) the mitigated case, in which 10 CFR 50.54 (hh)(2) mitigation strategies were assumed to be successful and (2) the unmitigated case, in which these mitigation strategies were assumed to be unsuccessful. Following the evaluation of these cases, the staff performed a limited scope human reliability analysis to estimate the likelihood of successful operator actions implementing 10 CFR 50.54(hh)(2) mitigation measures to prevent fuel damage. Key assumptions made in this limited scope human reliability analysis are that (1) post-earthquake onsite portable mitigation equipment required by 10 CFR 50.54(hh)(2) was available, (2) minimum plant staffing was available for implementing SFP mitigation, and (3) operators had access to areas needed to implement mitigation measures. The study considered scenarios in which some preplanned and improvised mitigating actions were either unsuccessful or not implemented before the analysis was terminated at 72 hours. For example, in addition to the 10 CFR 50.54(hh)(2) mitigation strategies, the site emergency response organization would request support from offsite response organizations to implement additional mitigating actions that are improvised, such as pumping water into the SFP using a fire truck. However, these additional mitigating actions were determined to be beyond the scope of the study.

Accident Scenario Selection

Previous risk studies had shown that earthquakes represent the dominant risk contributor for SFPs. Therefore, to deliberately challenge the integrity of the SFP, the accident initiator for this study was a beyond-design-basis earthquake with ground motion (0.7g peak ground acceleration) stronger than the maximum earthquake reasonably expected to occur for the reference plant. An earthquake of this severity was estimated to occur about once every 60,000 years.

The SFP accident scenarios evaluated in this study were developed for a single operating cycle. However, the conditions of the SFP change throughout an operating cycle. For example, the SFP can change from being an isolated pool to being hydraulically connected to the reactor vessel (e.g., during refueling operations), or spent fuel can be moved around within the SFP during a cycle to satisfy regulatory requirements with respect to criticality or heat distribution. Such changes affect the consequences of a postulated accident. Therefore, for this study, the

[31] A limited sensitivity analysis of a 1x8 spent fuel configuration and a uniform configuration was also performed to better understand the potential effects of plausible alternative SFP configurations on results and insights.

continual changes that occur during a single operating cycle were discretized into discrete quasi-steady snapshots referred to as operating cycle phases (OCPs). Since the number of OCPs has a roughly linear scaling effect on the number of MELCOR analyses required, the study defined in terms of the minimum number that most accurately represented pool-reactor configurations (i.e., whether the SFP is connected to the reactor), spent fuel loading configurations, and decay heat levels. Five OCPs were identified based on the timing of fuel movement, key changes in pool-reactor configuration, and peak assembly and whole pool decay heat curves, as listed in Table H-25. Note that, while the beyond-design-basis earthquake described above is equally likely to happen throughout an entire operating cycle, the conditional probability of it occurring during a given OCP is the length of time in an OCP divided by the duration of the entire operating cycle (i.e., fraction of time in each OCP).

Table H-25 Operating Cycle Phase Descriptions

OCP No.	OCP Description	OCP Time Duration (days)	% of Total Operating Cycle	Pool-Reactor Configuration*
1	Defueling of reactor core (~1/3 core)	2–8	0.9	Refueling
2	Reactor testing, maintenance, inspection and refueling	8–25	2.4	Refueling
3	Highest decay power portion of non-outage period	25–60	5	Unconnected
4	Next highest decay power portion of non-outage period	60–240	25.7	Unconnected
5	Remainder of operating cycle	240–700; 0–2	66	Unconnected

*Note: The “refueling” pool-reactor configuration refers to the configuration in which the SFP and the reactor are hydraulically connected. During other stages of the operating cycle, the SFP and reactor are not connected.

As part of scenario development, the study also considered onsite mitigation and offsite support. It treated onsite mitigation by modeling two cases, successful and unsuccessful mitigation, for each scenario. Successful mitigation occurred when mitigative actions required by 10 CFR 50.54(hh)(2) were successfully deployed, additional onsite capabilities were used to extend the use of the mitigation equipment, and arrival of offsite resources allowed the mitigative equipment to be used until onsite capabilities could be recovered. Unsuccessful mitigation occurred when none of the onsite mitigative actions were successful for an extended period. Offsite support was treated using the following assumptions:

- Offsite support arrives within 24 hours.
- Actions are planned, and equipment is staged within 48 hours.
- The accident progression analysis is truncated if the fuel is not uncovered and the pool can be refilled by 48 hours with an injection rate of 500 gallons per minute.
- If the above mitigation actions are unsuccessful, the sequence is run to 72 hours.

To develop accident scenarios, the NRC made several key assumptions based on structural analyses, including (1) all offsite and onsite alternating current power is lost as a result of the seismic event, (2) direct current power may be lost, (3) 10 CFR 50.54(hh)(2) equipment, when credited, is available for the duration of the event, (4) tearing of the SFP liner is possible, and (5) there is no failure of penetrations. Based on these and other assumptions, the NRC developed six accident cases for each OCP using a combination of zero, small, and moderate leakage damage states with successful and unsuccessful mitigation actions taken for each leakage scenario. The staff used these accident cases for both high- and low-density loading configurations, as summarized in Table H-26.

Table H-26 Scenario Descriptions for a Given Operating Cycle Phase

Case No.	Scenario Characteristics	
	SFP Leakage Rate	Mitigation?
1	None	Yes
2		No
3	Small	Yes
4		No
5	Moderate	Yes
6		No

MELCOR Severe Accident Progression and Source Term Analyses

Analysts used the MELCOR code (Version 1.8.6) to model severe accident progression for the scenarios described in the previous section. Enclosure H-1, "Description of Analytical Tools and Capabilities," to this appendix describes the MELCOR code. The code was ideal for modeling accident progression for SFPs because SFP models had already been developed and validated, and it was also capable of modeling in-building transport/retention and radionuclide release, the latter of which was a key input for subsequent accident consequence analysis modeling using the MELCOR Accident Consequence Code System (MACCS).

To facilitate modeling of the SFP for BWR fuel assemblies, the staff used a recently developed rack component for improved spent fuel rack modeling and an oxidation kinetics model. These two additions to MELCOR enabled the evaluation of two types of SFP accidents: a partial loss-of-coolant inventory or boiloff accident, and a complete loss-of-coolant inventory accident. A partial loss-of-coolant inventory or boiloff accident could involve no or late uncovery of the bottom of the racks, and boiloff of the coolant could ultimately lead to hydrogen combustion. A complete loss-of-coolant accident occurs when the bottom of the racks is uncovered, leading to air oxidation of the cladding and enhanced ruthenium release.

The staff used the radionuclide package in MELCOR to model the release and transport of fission product vapors and aerosols. It tracks radionuclides by combining them into material classes, which are groups of elements with similar chemical and transport behavior. The SFP MELCOR model includes 15 default material classes and 2 user-defined classes that can model cesium iodide and cesium molybdate behavior. This study modified the default cesium, iodine, and molybdenum radionuclide classes to accommodate new insights obtained from the Phebus experimental program.[32] In addition, the staff developed a new ruthenium release model in which it adjusted the default vapor pressure parameters for the ruthenium material class to match the ruthenium dioxide vapor pressure at 2,200 K. However, it only used this latter model in scenarios involving rapid draindown (i.e., moderate leak rates) in the SFP. All scenarios applied a 5 percent gap release criterion.

[32] The PHEBUS Fission Products international research program took place between 1988 and 2010. Its purpose was to improve the understanding of the phenomena occurring during a core meltdown accident in a light-water reactor and to reduce uncertainties in calculated radionuclide releases for reactor safety evaluations that model core meltdown accidents.

The decay heat and radionuclide packages were used to calculate the fission product inventory and specific decay power for 29 elemental groups; the specific elemental decay power is compiled as a function of time after shutdown. Because these packages were originally designed for reactor accident progression analyses, the shutdown time for each assembly is the same. Unlike the case for reactor accidents, SFP accidents involve fuel assemblies with multiple shutdown times. To address this discrepancy, a scaling procedure in MELCOR enabled the use of batch-average decay heat results. Each batch also used a post-processing routine with MELCOR-predicted release fractions and actual inventories. Lastly, to map the calculated releases from MELCOR to the MACCS[33] code for accident consequence analyses, the MELCOR input file was modified to enable tracking of fission product releases from each ring, or collection of assemblies in the MELCOR radial nodalization, as well as the subsequent releases to the environment.

To calculate the above mentioned radionuclides and decay heats, the reference plant's utility provided information for all assemblies that had been discharged from the reference plant to the SFP over 18 cycles. From this information, the actual analysis basis for the high-density SFP inventory was 3,055 assemblies, based on the SFP capacity of 3,819 assemblies minus 764 assemblies to accommodate a full core offload capability. Although the utility provided data for 18 discharge cycles, this study only included cycles 7–18, since these cycles provided the requisite target inventory (3,055 assemblies). For the burnup analysis, the ORIGEN code simulated the irradiation and decay history for each of the 3,055 assemblies. In this case, the assemblies were each decayed to a reference date, which was the end of the last cycle (18), and the resulting inventories were combined into groups for analysis. These analysis groups were additionally decayed to determine assembly activities and decay heat power to simulate cooling of the discharged fuel after reactor shutdown. The assemblies were then placed into six groups according to the cycle in which they were discharged. The benefit of grouping these assemblies in this manner is that it facilitated the use of the data for analyses of low-density SFP configurations in which assemblies that had been cooled for more than 5 years were removed.

Description of SFP MELCOR Models

The SFP for the reference plant is located on the refueling floor of the reactor building. In one corner of the SFP is a cask area. At the bottom of the SFP, high-density SFP racks are located to store the SFP. During operation, these racks are covered with approximately 23 feet of water to provide radiation shielding. Each rack is rectilinear in shape and comes in nine different sizes, and a total of 3,819 storage locations are located in the pool. Each stainless-steel rack includes cell assemblies, a baseplate with flow-through holes, and base support assemblies.

For the entire SFP model, MELCOR used a series of control volumes for regions at the top and bottom of the SFP (see Figures 39 and 40 in NUREG-2161). The region at the bottom of the SFP containing the empty and loaded spent fuel storage racks was more finely divided into several control volumes to enable detailed analyses of all 3,819 storage locations for high- and low-density configurations. The BWR assembly canisters were modeled using the MELCOR canister component. In addition to the detailed SFP model, the staff used a simplified reactor building model consisting solely of the refueling room, since the bulk of the reactor building

[33] At the time of this analysis, the MACCS code was called the "MACCS2" code, a leftover notation from the time that the original MACCS code was substantially upgraded to Version 2. Since then, the staff has referred to the code as the "MACCS" code and notes the version number of the code used in a particular analysis, since code development and maintenance continues.

components do not play a significant role in SFP accidents. The refueling room was modeled using a single control volume in MELCOR, which accounted for nominal reactor building leakage and simulated overpressure failure flowpaths.

To model reactor outages in which the SFP and the reactor are hydraulically connected (i.e., OCP1 and OCP2), a single control volume represented the reactor well and separator/dryer pool. This control volume was then connected to the spent fuel model described above for the analyses. For each OCP, the assembly layout was also modified to account for assembly offloads for both the high- and low-density loadings.

MELCOR Accident Progression Analysis Results and Source Terms

The MELCOR analyses of the six cases per OCP and illustrated in Table H-26 revealed that four classes of scenarios did not lead to a release:

- boiloff scenarios with no SFP leaks

- mitigated scenarios for small leaks

- unmitigated scenarios in late phases (OCP4, OCP5)

- mitigated moderate leak scenarios in OCP2, OCP3, OCP4, and OCP5

For the boiloff scenarios, a simplified MELCOR model in which all assemblies are combined in only two rings (collections of assemblies) that represent the fuel and empty cells was used to estimate the pool heatup and water level drop. The study used the thermal-hydraulic models in MELCOR, and the simplified model for boiloff, to evaluate sets of both low-density and high-density cases. For both sets, no release occurred because the water level never dropped below the top of the SFP racks. If boiloff of the coolant below the top of the SFP racks had occurred, it could have led to steam generation, oxidation of the cladding, hydrogen production, and possibly hydrogen combustion and release of radionuclides. Similarly, none of the mitigated scenarios for small leaks led to release during any OCP because the rate of water injection (500 gallons per minute) as a mitigative action ensured that the fuel never became uncovered or overheated.

The results of MELCOR analyses of the unmitigated scenarios in OCP4 and OCP5 indicated that, although there was fuel heatup in both high- and low-density configurations after the rack baseplate was uncovered, there was no release because the total decay heat of the assemblies in these stages was at least 37 to 48 percent lower than the total decay heat of assemblies in OCP3, and natural circulation was sufficient to slow down the rate of fuel heatup to the point at which the fuel failure could occur.

For moderate leaks, mitigation involved spray activation for outage phases OCP1 and OCP2, and direct injection for post-outage phases OCP3, OCP4, and OCP5. The results of analyses of moderate leaks during phase OCP2 indicated that no releases occurred from various heat transfer mechanisms. Since the unmitigated scenarios for phases OCP3, OCP4, and OCP5 led to no release, the study only evaluated the results of the high-density moderate leak scenario for phase OCP3 (with and without spray flow turned on). The staff determined that modeling the mitigation of moderate leak scenarios with and without the spray mechanism activated led to no release of radionuclides because the fuel clad temperature never surpassed 900 degrees

Celsius (C) (1,652 degrees Fahrenheit (F)), at which point gap release would begin to occur. A key observation was that these results underscored the importance of natural circulation of air through the racks for heat removal to help keep the fuel clad temperatures below the gap release temperature. The study also modeled the moderate leak scenario for OCP3, assuming an additional 3-hour delayed activation of the spray for a spray activation time of 6 hours after the leak occurs. In this case, it was shown that the maximum clad temperature reached just under 627 degrees C (1,160 degrees F) after 6 hours, at which point the activated spray was sufficient to keep the fuel clad well below the gap release temperature of 900 degrees C (1,652 degrees F).

The 14 scenarios that led to release of radionuclides can be categorized as follows:

- unmitigated small leaks in OCP1, OCP2, and OCP3, in both high- and low-density configurations

- unmitigated moderate leaks in OCP1, OCP2, and OCP3, in both high- and low-density configurations

- mitigated moderate leak in OCP1 in both high- and low-density configurations

Tables H-27 and H-28 summarize the release characteristics for the 14 scenarios that led to a release of radionuclides.

Table H-27 Summary of Release Results for High-Density Configurations

High-Density Case No.	Scenario Characteristics		Release Characteristics			
	SFP Leakage	50.54(hh)(2) Equipment?	Cesium Release at 72 hours	Cs-137 Released (MCi)	Iodine Release at 72 hours	I-131 Released (MCi)
OCP1	Small	No	0.6%	0.33	3.5%	0.27
	Moderate	Yes	0.5%	0.26	5.0%	0.39
	Moderate	No	1.5%	0.8	2.1%	0.16
OCP2	Small	No	17.1%	7.90	17.1%	1.91
	Moderate	No	1.6%	0.73	2.0%	0.22
OCP3	Small	No	42.0%	24.20	51.2%	0.73
	Moderate	No	0.7%	0.39	0.7%	0.01

Table H-28 Summary of Release Results for Low-Density Configurations

Low-Density Case No.	Scenario Characteristics		Release Characteristics			
	SFP Leakage	50.54(hh)(2) Equipment?	Cesium Release at 72 hours	Cs-137 Released (MCi)	Iodine Release at 72 hours	I-131 Released (MCi)
OCP1	Small	No	3.1%	0.33	4.6%	0.36
	Moderate	Yes	1.8%	0.19	7.0%	0.55
	Moderate	No	0.5%	0.05	1.7%	0.13
OCP2	Small	No	1.7%	0.28	3.3%	0.37
	Moderate	No	0.4%	0.07	0.7%	0.08
OCP3	Small	No	0.6%	0.10	1.2%	0.02
	Moderate	No	0.1%	0.02	0.2%	0.00

Unmitigated moderate leaks for high-density configurations in OCP1, OCP2, and OCP3 did not lead to hydrogen deflagration, and the releases were relatively low since oxygen depletion limited clad oxidation and fuel heatup. Similarly, none of the scenarios for the low-density configurations led to hydrogen deflagration, and the release fractions were typically low and comparable to the analogous scenario for the high-density loading configuration. One exception to this trend is the low-density OCP1 scenario for mitigated moderate leaks. In this case, the low-density case has slightly higher releases than the high-density cases because there was higher and faster heatup of the most recently discharged assemblies in the low-density cases. The higher initial fuel temperatures in the low-density case led to slightly higher releases. Notably, the highest release fractions for cesium and iodine were observed for scenarios that led to hydrogen combustion; namely, unmitigated small leaks for high-density configurations in OCP2 and OCP3.

The release data in the tables above were used as input for the accident consequence analyses, as described in the following section.

MACCS Consequence Analyses

Based on results from the MELCOR modeling of SFP accident progression scenarios, the staff used Version 2 of the MELCOR Accident Consequence Code System (MACCS, Revision 3.7.0) to model offsite consequence analyses. MACCS can evaluate the impacts of atmospheric releases of radioactive aerosols and vapors on human health and on the environment by using site-specific weather conditions, population data, and evacuation plans. Quantification of the effects of offsite radioactive releases on human health is accomplished by modeling and evaluating the relevant dose pathways; namely, cloudshine, inhalation, groundshine, and ingestion. Enclosure H-1 to this appendix describes the MACCS suite of codes.

A source term definition was created for each accident consequence evaluation as described below. The ORIGEN code calculated the activity levels of the different radionuclides of the fuel in the SFP, while the plume characteristics—including chemical group release rates, aerosol size distributions, density, and mass flow rates—were obtained from the MELCOR analyses described in the previous section. The 14 MELCOR sequences that led to release (see Tables H-27 and H-28 above) were binned by their cesium (Cs)-137 and iodine (I)-131 release activities to lessen the computational cost of the MACCS calculations. Sequences were first grouped into three bins based on their Cs-137 release activities (i.e., 0–0.25, 0.25–0.55, and greater than 0.55 megacuries (MCi) of Cs-137 released) because Cs-137 is the most significant contributor to long-term consequences and groundshine dose. The sequences were then binned based on I-131 release (i.e., 0–0.5, 0.5–5, and greater than 5 MCi of I-131 released) because I-131 is a good indicator for short-lived radionuclides that may be released from recently discharged fuel. In this manner, the 14 release sequences were ultimately binned into nine radiological release categories (RCs), with only four RCs containing at least two release sequences. The staff chose one sequence from each of the four RCs to represent the entire RC except for RC33. The study analyzed both release sequences in RC3 because these release sequences had the highest releases of all sequences. The binning of the 14 MELCOR sequences that led to release is illustrated in Tables H-29 and H-30 for high-density and low-density loading cases with and without mitigation. The sequences that were selected for further analysis are indicated in Tables H-29 and H-30 with bold text for emphasis.

Table H-29 Binning of MELCOR Release Sequences into Release Categories for High-Density Configurations

High-Density Case No.	Scenario Characteristics		Release Characteristics			
	SFP Leakage	50.54(hh)(2) Equipment Deployed	Cs-137 Released (MCi)	I-131 Released (MCi)	Release Category	Sequence Analyzed in MACCS
OCP1	**Small****	**No**	**0.33**	**0.27**	**RC12**	**Yes**
	Moderate	Yes	0.26	0.39	RC12	No
	Moderate	No	0.8	0.16	RC21	No
OCP2	**Small**	**No**	**7.90**	**1.91**	**RC33**	**Yes***
	Moderate	**No**	**0.73**	**0.22**	**RC21**	**Yes**
OCP3	**Small**	**No**	**24.20**	**0.73**	**RC33**	**Yes***
	Moderate	No	0.39	0.01	RC11	No

*The release scenarios for both sequences in RC33 were evaluated in MACCS because of the comparatively higher releases compared to other scenarios.
**The sequences that were selected for further analysis are indicated with bold font.

Table H-30 Binning of MELCOR Release Sequences into Release Categories for Low-Density Configurations

Low-Density Case No.	Scenario Characteristics		Release Characteristics			
	SFP Leakage	50.54(hh)(2) Equipment Deployed	Cs-137 Released (MCi)	I-131 Released (MCi)	Release Category	Sequence Analyzed in MACCS
OCP1	Small	No	0.33	0.36	RC12	No
	Moderate	Yes	0.19	0.55	RC12	No
	Moderate	No	0.05	0.13	RC11	No
OCP2	Small	No	0.28	0.37	RC12	No
	Moderate	No	0.07	0.08	RC11	No
OCP3	**Small**	**No**	**0.10**	**0.02**	**RC11**	**Yes**
	Moderate	No	0.02	0.00	RC11	No

*The sequence that was selected for further analysis is indicated with bold font.

The release data described above were used in MACCS for subsequent atmospheric transport and dispersion modeling; exposure, dosimetry, and health effects modeling; emergency response modeling; and long-term protective action modeling, as described in the next section.

MACCS Model Descriptions

Atmospheric Transport and Dispersion Modeling

The MACCS straight-line Gaussian plume segment dispersion model was used to model the atmospheric transport and dispersion of radionuclides released for a given accident scenario. The study divided radionuclides released into the atmosphere into plume segments that are 1 hour or less to match the resolution of the dispersion models to that of the weather data. In addition, the aerosol size distributions obtained from MELCOR, combined with the aerosol velocity data obtained from NUREG/CR-7161, “Synthesis of Distributions Representing Important Non-Site-Specific Parameters in Off-Site Consequence Analyses,” issued April 2013, were used to model deposition rates of aerosols from the plume to the ground.

One year of hourly meteorological data from onsite meteorological tower observations documented in NUREG-1935, “State-of-the-Art Reactor Consequence Analyses (SOARCA) Report,” was used for atmospheric modeling in this study. Specifically, the study used meteorological data from the year 2006 at the reference plant site was used. Since the exact weather conditions for a potential future accident are unknown, MACCS accounts for weather variability by analyzing a statistically significant set of weather trials. In this way, the modeled results are an ensemble that represents the full spectrum of meteorological conditions. The nonuniform weather binning strategy used to sample sets of weather data is based on the approach used in NUREG/CR-7009, “MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project,” issued August 2014.”

Exposure, Dosimetry, and Health Effects Modeling

Groundshine, cloudshine, inhalation, and ingestion are exposure pathways considered in MACCS to calculate population dose and health effects. In general, food ingestion parameters in NUREG/CR-6613, Volume 1, “Code Manual for MACCS2: User’s Guide,” issued May 1998, were used to calculate ingestion dose. Shielding factors applied to evacuation, normal activity, and sheltering for each dose pathway were obtained from NUREG/CR-7009.

The Federal Guidance Report 13, “Cancer Risk Coefficients for Environmental Exposures to Radionuclides,” issued September 1999, provided the dose coefficients, risk factors, and relative biological effectiveness. As implemented in MACCS, the Federal Guidance Report 13 dose coefficients along with the dose and dose rate effectiveness factors were incorporated in the dose response modeling for the early phase for doses less than 20 rem and in the long-term phase of the offsite consequences. The risk factors were implemented in MACCS for seven organ-specific cancers, as well as residual cancers that were not accounted for directly. NUREG/CR-7161 provided parameters related to health effects, as well as other non-site-specific data used for consequence analysis.

The NRC used SECPOP2000 to create a MACCS site file containing population and economic data for 16 compass sectors. The site file was then interpolated onto a 64-compass sector grid to improve spatial resolution for the consequence analysis. Site population data were extrapolated to the year 2011 using census data from the year 2000 and a multiplier of 1.1051 from the U.S. Census Bureau to account for the average population growth in the United States between 2000 and 2011. Similarly, economic values from the SECPOP2000 database, whose values are based on year 2002 economic data, were scaled by 1.250 derived, based on the consumer price index to account for price escalation (i.e., increasing value of the dollar) between 2002 and 2011.

Emergency Response Modeling

The MACCS models for the emergency phase, which is the 7-day period following the start of a release, calculated the dose and associated health effects to the public as well as the effects of emergency preparedness actions that protect the public. To model emergency response the staff developed three evacuation models based on whether 4-day dose projections were expected to exceed 1 rem for a member of the public, at which point the protective action guideline (PAG) was considered to be exceeded—(1) a small projected dose that does not exceed the PAG at the emergency planning zone (EPZ), (2) a large projected dose (within 48 hours) that exceeds the PAG at the EPZ, and (3) a large projected dose (within 24 hours) that exceeds the PAG at the EPZ. For each model, specific protective actions (e.g., general public evacuation, hotspot relocation, shadow relocation) were included for populations within and beyond the EPZ. To model population evacuation in these models, the population was divided into cohorts, which are population groups that move differently from other groups. The cohorts were loaded onto the roadway network at a specified time, and a set of speed values were applied per cohort for the early, middle, and late periods of the evacuation. The last 10 percent of the population to evacuate (i.e., the evacuation tail) was modeled as a separate cohort. For residents within the EPZ, the MACCS potassium iodide model used in the analysis assumes that potassium iodide would only be distributed within the EPZ, and 50 percent of the population within the EPZ would have access to and take it as directed.

Long-term Protective Action Modeling

MACCS was also used to model the long-term protective action phase (i.e., the 50-year period following the 7-day emergency phase). Three protective actions were modeled for contaminated land during the long-term phase: interdiction, decontamination, and condemnation. In the MACCS model, interdiction and condemnation are defined in terms of habitability. Interdiction is a temporary relocation during which land contamination levels are reduced by decontamination, natural weathering, and radioactive decay to restore habitability. If contamination levels cannot be adequately reduced to restore habitability within 30 years, the land is considered condemned, and the population is modeled not to return during the long-term phase (i.e., permanently relocated). Based on the location of the reference plant in Pennsylvania, this study used a habitability criterion of 500 millirem (mrem) per year beginning in the first year. Two levels of decontamination with decontamination factors of 3 and 15 were modeled for a 1-year timespan. The cost of decontamination during this period was determined using values in NUREG/CR-7009.

This study also considered land suitable for farming (farmability). Values used to define farmability were taken from NUREG-1150, “Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants,” issued December 1990.” Agricultural land with contamination levels in excess of the farmability criteria was considered unfarmable, and no farming was allowed until the farmability criteria were satisfied.

MACCS Consequence Analysis Results

Table H-31 summarizes the mean reduction in offsite consequence results in terms of averted population dose (person-rem) and averted economic costs (2012 dollars) associated with implementing Option 2 (expedited spent fuel transfer to achieve low-density SFP storage). The reported consequence metrics represent averted consequences that were calculated by taking the difference between consequences for Option 1 (regulatory baseline) and consequences for Option 2.

Table H-31 Mean Reduction in Offsite Consequence Results Associated with Option 2

Consequence Metric[a]	Best Estimate	Low Estimate	High Estimate
Averted 50-mile Population Dose (person-rem)	124	60	1260
Averted 50-mile Economic Costs (2012 dollars)	$723,300	$1,073,300	$4,587,800

[a] The reported consequence metrics represent averted consequences that were calculated by taking the difference between consequences for Option 1 (regulatory baseline) and consequences for Option 2 (expedited spent fuel transfer to achieve low-density SFP storage).

The consequence metrics for population dose and economic costs can vary significantly with the criterion used to measure or estimate the level of land contamination and to inform decisions about when to allow relocated populations to return to contaminated land areas. The offsite consequence analysis performed in support of SECY-13-0112 and NUREG-2161 used three PAG levels based on annual dose to calculate the estimates of averted population dose and averted economic costs within 50 miles: (1) the U.S. Environmental Protection Agency (EPA) intermediate phase PAG level of 2 rem in the first year, and 500 mrem annually thereafter, was used to calculate the best estimate, (2) the more stringent Pennsylvania PAG level of 500 mrem annually starting with the first year was used to calculate the low estimate, and (3) the less stringent 2 rem annually was used to calculate the high estimate. The analysis calculated all estimates assuming a remaining licensed term of 22 years (until 2034) for the reference plant and using the reference site's offsite population density within a 50-mile radius from the site (approximately 722 people per square mile).

The study included a limited treatment of uncertainty by describing results for a range of sensitivity analyses performed to evaluate the effect of certain assumptions on results and insights. Factors addressed in these sensitivity analyses included the following:

- using a more favorable 1x8 fuel assembly pattern
- using an unfavorable uniform fuel assembly pattern
- radiative heat transfer
- hydrogen combustion ignition criterion
- occurrence of concurrent events involving the reactor or multiunit events
- molten core-concrete interaction
- alternative accident scenario truncation times
- effects of reactor building leakage on hydrogen combustion and accident progression

Risk Evaluation

This study was a limited scope consequence analysis supplemented with probabilistic insights to provide additional context and perspectives about the relative likelihood of events and consequences. This analysis considered the following as examples of probabilistic insights:

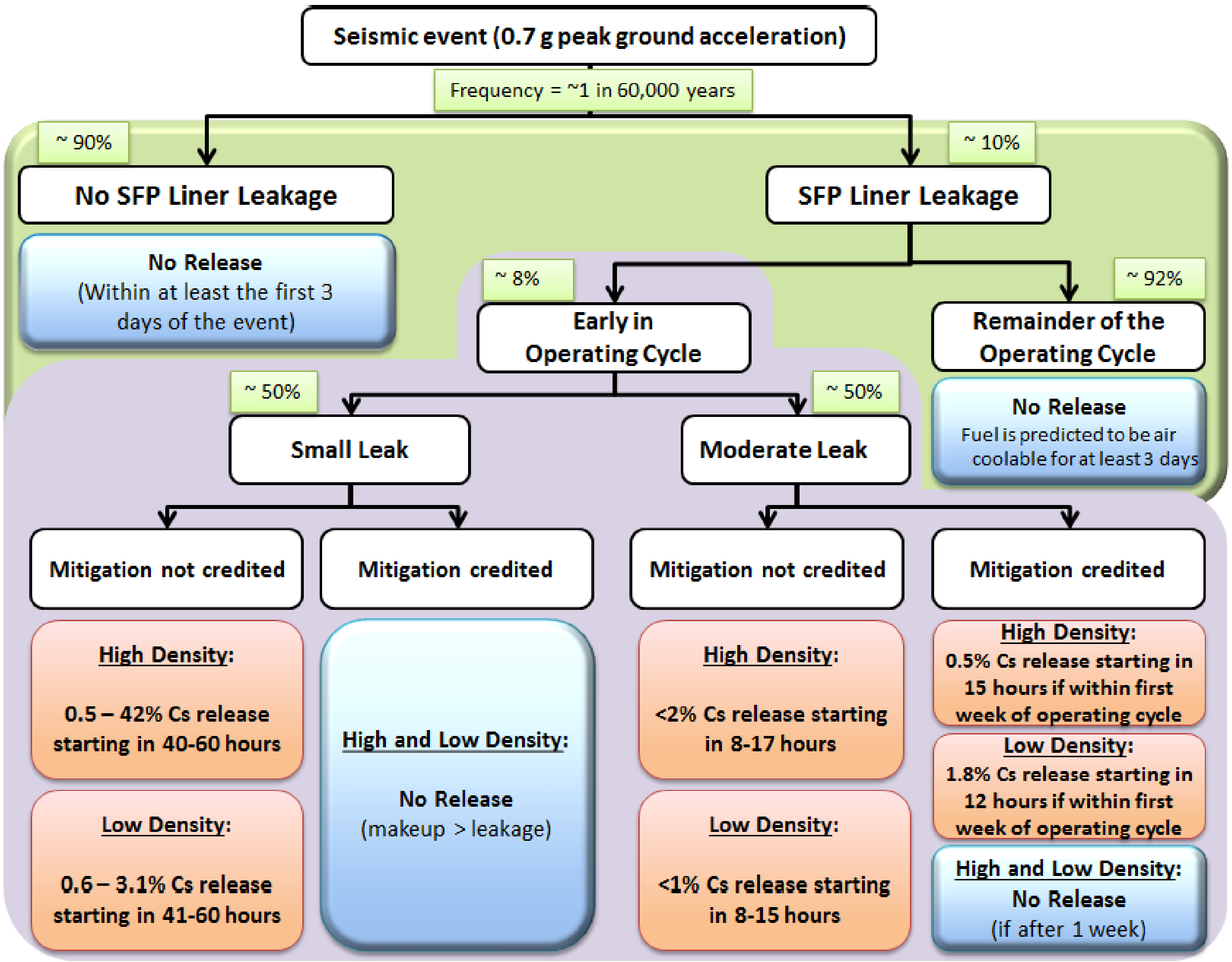
- risk information from past studies for accident scenario selection
- initiating event frequency information
- initiating event timing effects (e.g., the relative likelihood of an event occurring during each OCP and the likely configurations incurred)
- relative likelihoods of damage state characteristics
- probabilistic consequence analysis to account for effects of statistical variability in offsite weather conditions on offsite radiological consequences

While these elements provided some of the benefits of PRA, this study did not perform several elements of a traditional PRA. The following are examples of traditional PRA elements that were excluded from this study:

- failure modes and effects analysis (except for certain structures, systems, or components specifically identified in the study)
- data analysis and component reliability estimation
- dependency analysis
- human reliability analysis as part of the accident progression and recovery (except the limited scope human reliability analysis that was performed as described above)
- system fault tree and accident sequence event tree development and quantification

Figure H-20 illustrates the conditional probability of SFP liner leakage and magnitude of release from the SFP—conditioned on the assumed occurrence of the beyond-design-basis earthquake considered in the study—for postulated accident scenarios that occur in different phases of the operating cycle. The figure shows the results for both the high-density and low-density loading configurations, as well as for the mitigated and unmitigated cases.

The inclusion of probabilistic insights allowed analysts to consider some aspects of likelihood but could not support making definitive statements about SFP risk. This study focused on a specific portion of the overall risk profile—SFP accidents caused by large seismic events between 0.5g and 1g. This study can therefore be used to corroborate or challenge the continued applicability of estimates for this part of the risk profile based on previous studies. In addition, since large seismic events have been shown in the past to be a dominant contributor to SFP risk, this comparison helps to predict whether a full-scope PRA would be expected to result in an overall decrease or increase in estimated risk. Therefore, the results of this study can draw supportable, but not definitive, conclusions about overall SFP risk.



Note: The low-density pool has about 1/3 of Cs-137 inventory compared to high-density pool. Early in the operating cycle refers to early time after shutdown.

Figure H-20 Conditional Probability of SFP Liner Leakage and SFP Release Magnitude

Cost-Benefit Analysis Results

Table H-32 summarizes the results of the quantitative cost-benefit analysis for the best estimate and low– and high-estimate cases for Option 2, documented in NUREG-2161, Appendix D. At the time this regulatory analysis was prepared, returns on investments were well below the 3 percent and 7 percent discount rates described in the Office of Management and Budget (OMB) Circular No. A-4, "Regulatory Analysis," dated September 17, 2003. A sensitivity analysis was performed using a 0 percent discount rate that produced undiscounted values in constant dollars. Although it was common practice to provide undiscounted values for costs and benefits for information purposes within regulatory analyses, it was not common practice to report such results as part of a sensitivity analysis. However, the staff chose to report the undiscounted costs and benefits as part of a sensitivity analysis for this regulatory analysis to account for current market trends and future predictions. Note that this enclosure[34] only discusses the calculation of public health and offsite property attributes, which is based on the detailed severe accident analysis using MELCOR and MACCS.

[34] Methods for calculating occupational health, onsite property, and implementation costs are discussed elsewhere in NUREG-2161.

In addition to the sensitivity analysis described above to evaluate the effect on results of using a 0 percent discount rate, the staff performed sensitivity analyses to account for the effect on the results of (1) using an alternative dollar per person-rem conversion factor ($4,000 per person-rem instead of $2,000 per person-rem), (2) extending the analysis of consequences beyond a 50-mile circular radius around the site, and (3) combining the effects of using the $4,000 per person-rem conversion factor and extending the analysis of consequences beyond 50 miles from the site. Tables H-32 and H-33 summarize the results of these sensitivity analyses.

As shown in Table H-33, requiring the expedited transfer of spent fuel from the SFP to dry cask storage to achieve low-density SFP storage at the reference plant did not achieve a positive net benefit for eight of the nine cases presented. The undiscounted high-estimate case—which reflects the costs and benefits at the time in which they are incurred with no present worth conversion and which assumes the least stringent habitability criterion—resulted in a positive net benefit of about $27.1 million. However, the other high-estimate cases resulted in negative net benefits of about ($10.6 million) and ($25.1 million), which differed from this case by adjusting future costs and benefits into 2012 dollars using 3 percent and 7 percent discount rates, respectively.

Table H-32 Summary of Benefits and Costs within 50 Miles for Option 2

Attribute	Best Estimate[a]			Low Estimate[a]			High Estimate[a]		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$247,700	$179,500	$124,600	$119,700	$86,700	$60,200	$2,520,000	$1,825,500	$1,267,000
Occupational Health (Accident)	$1,300	$900	$700	$700	$500	$300	$21,300	$15,400	$10,700
Offsite Property	$723,300	$524,000	$363,700	$1,073,300	$777,500	$539,700	$4,587,800	$3,323,400	$2,306,700
Onsite Property	$10,400	$6,900	$4,300	$4,480	$2,950	$1,830	$378,600	$249,600	$155,800
Total Benefits	$982,700	$711,300	$493,300	$1,198,200	$867,700	$602,000	$7,507,700	$5,413,900	$3,740,200
Occupational Health (Routine)	($9,000)[c]	($24,000)	($27,000)	($9,000)	($24,000)	($27,000)	($9,000)	($24,000)	($27,000)
Industry Implementation	($15,660,000)	($41,820,000)	($46,770,000)	($15,660,000)	($41,820,000)	($46,770,000)	($15,660,000)	($41,820,000)	($46,770,000)
Industry Operation	($730,000)	($252,000)	($64,000)	($730,000)	($252,000)	($64,000)	($730,000)	($252,000)	($64,000)
NRC Implementation	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]
NRC Operation	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]	NC[b]
Total Costs	($16,399,000)	($42,096,000)	($46,861,000)	($16,399,000)	($42,096,000)	($46,861,000)	($16,399,000)	($42,096,000)	($46,861,000)
Net Benefit	($15,416,000)	($41,385,000)	($46,368,000)	($15,200,800)	($41,228,300)	($46,259,000)	($8,891,300)	($36,682,100)	($43,120,800)

[a] Discounted net present value (NPV) results are expressed in 2012 dollars. Undiscounted results are expressed in constant dollars.
[b] NC: Not calculated
[c] Negative values are shown using parentheses (e.g., negative $9,000 is displayed as ($9,000)).

Table H-33 Combined Effect of $4,000 per Person-Rem Conversion Factor and Consequences Beyond 50 Miles for Option 2

Attribute	Best Estimate[a]			Low Estimate[a]			High Estimate[a]		
	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV	Undiscounted	3% NPV	7% NPV
Public Health (Accident)	$3,566,900	$2,583,800	$1,793,400	$2,162,500	$1,566,500	$1,087,300	$31,471,600	$22,798,200	$15,823,400
Occupational Health (Accident)	$2,500	$1,900	$1,400	$1,300	$1,000	$700	$42,700	$30,900	$21,400
Offsite Property	$2,139,300	$1,549,700	$1,075,600	$4,968,300	$3,599,100	$2,498,000	$11,586,600	$8,393,400	$5,825,500
Onsite Property	$10,400	$6,900	$4,300	$4,680	$3,150	$2,030	$378,600	$249,600	$155,800
Total Benefits	$5,719,100	$4,142,300	$2,874,700	$7,136,800	$5,169,800	$3,588,000	$43,479,500	$31,472,100	$21,826,100
Occupational Health (Routine)	($18,000)[c]	($49,000)	($54,000)	($18,000)	($49,000)	($54,000)	($18,000)	($49,000)	($54,000)
Industry Implementation	($15,660,000)	($41,820,000)	($46,770,000)	($15,660,000)	($41,820,000)	($46,770,000)	($15,660,000)	($41,820,000)	($46,770,000)
Industry Operation	($730,000)	($252,000)	($64,000)	($730,000)	($252,000)	($64,000)	($730,000)	($252,000)	($64,000)
NRC Implementation	NC[b]	NC	NC	NC	NC	NC	NC	NC	NC
NRC Operation	NC	NC	NC	NC	NC	NC	NC	NC	NC
Total Costs	($16,408,000)	($42,121,000)	($46,888,000)	($16,408,000)	($42,121,000)	($46,888,000)	($16,408,000)	($42,121,000)	($46,888,000)
Net Benefit	($10,689,000)	($37,979,000)	($44,013,000)	($9,271,200)	($36,951,200)	($43,300,000)	$27,071,500	($10,648,900)	($25,061,900)

[a] Discounted net present value (NPV) results are expressed in 2012 dollars. Undiscounted results are expressed in constant dollars.
[b] NC: Not calculated
[c] Negative values are shown using parentheses (e.g., negative $18,000 is displayed as ($18,000)).

Summary and Conclusion

Table H-32 shows that requiring the expedited transfer of spent fuel from the SFP to dry cask storage to achieve low-density SFP storage does not achieve a cost-beneficial increase in public health and safety for the reference plant using the current regulatory framework. In addition, three sensitivity analyses (Table H-33) also showed that the regulatory alternative represented by Option 2 was not cost-beneficial for any cases in which costs and benefits incurred in the future were discounted to their present worth using 3 percent and 7 percent discount rates consistent with OMB guidance. Moreover, the staff identified other considerations that would further reduce the quantified benefits, thereby making Option 2 even less justifiable. These other considerations included (1) the costs and risks associated with the handling and movement of spent fuel casks in the reactor building, (2) the post-Fukushima mitigation strategies required under Order EA-12-049 and the reliable SFP instrumentation required under Order EA-12-051, which significantly enhance the likelihood of successful mitigation, and thereby reduce the conditional probability of radiological release, and (3) the possibility of other favorable SFP loading configurations.

Based on its quantitative cost-benefit analysis, the staff concluded that the added costs involved in expediting the transfer of spent fuel from the SFP to dry cask storage to achieve low-density SFP storage at the reference plant were not warranted. In addition, based on the results of its safety goal evaluation, the staff concluded that this regulatory alternative could not result in a substantial increase in overall protection of public health and safety. Together, these analyses indicated that—for the reference plant—requiring the expedited transfer of spent fuel from the SFP to dry cask storage to achieve low-density SFP storage was not justified.

However through its analysis, the staff discovered that an alternative 1x8 high-density fuel configuration may have significantly lower implementation costs and potentially similar benefits to the low-density configuration. The staff therefore recommended that this alternative—in addition to other possible SFP loading configurations—be evaluated further as part of a subsequent regulatory analysis that would be performed to more broadly assess whether any significant safety benefits (or detriments) would occur from requiring expedited spent fuel transfer from SFPs to dry storage casks for the range of SFP designs at existing and new (future) nuclear power plants. In SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012, the staff provided a five-step plan to evaluate whether regulatory action is warranted for the expedited transfer of spent fuel from SFPs into dry cask storage. Enclosure H-6 to this appendix summarizes the subsequent regulatory analysis that addresses this issue and that is documented in COMSECY-13-0030.

Commission's Response to the Staff's Analysis and Recommendations

The staff provided SECY-13-0112 to the Commission as an information paper instead of a notation vote paper. Therefore, the Commission did not vote on the staff's analysis and its recommendations provided therein. However, after receiving the Tier 3 program plan documented in SECY-12-0095, the Commission directed the staff in several related SRMs. Enclosure H-6 to this appendix summarizes this Commission direction.

ENCLOSURE H-6: SUMMARY OF DETAILED ANALYSES IN COMSECY-13-0030, ENCLOSURE 1, "REGULATORY ANALYSIS FOR JAPAN LESSONS-LEARNED TIER 3 ISSUE ON EXPEDITED TRANSFER OF SPENT FUEL"

This enclosure summarizes the U.S. Nuclear Regulatory Commission (NRC) staff's regulatory analyses of whether expedited transfer of spent fuel to dry cask storage is warranted, as documented in COMSECY-13-0030, "Staff Evaluation and Recommendation, Enclosure 1, "Regulatory Analysis for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated November 12, 2013." These analyses used insights from and expanded upon the staff's previous evaluations described in NUREG-2161, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," issued September 2014, and SECY-13-0112, Enclosure 1, "Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor," dated October 9, 2014, and summarized in Enclosure H-5, "Summary of Detailed Analyses for SECY-13-0112 and NUREG-2161, 'Consequence Study of a Beyond-Design-Basis Earthquake Affecting the Spent Fuel Pool for a U.S. Mark I Boiling-Water Reactor,'" of this appendix. As such, this enclosure should be considered with the content of Enclosure H-5.

Problem Statement and Regulatory Objectives

The March 11, 2011, Great Tōhoku earthquake and subsequent tsunami in Japan caused extensive damage to the nuclear reactors at the Fukushima Dai-ichi nuclear power plant. Although the spent fuel pools (SFPs) and spent fuel assemblies remained intact, the event led to questions about the safe storage of spent fuel in SFPs and whether expedited transfer of spent fuel to dry cask storage was necessary. The event also generated increased interest in understanding the consequences of SFP accidents. On March 23, 2011, the NRC, in response to the accident at Fukushima Dai-ichi, on March 23, 2011, the NRC established a Near-Term Task Force (NTTF) to determine whether the NRC should make any near- or long-term improvements to its regulatory system, based on insights obtained from the Fukushima Dai-ichi accident. Nearly 4 months later, the NTTF provided its recommendations for regulatory improvements, including those to enhance SFP safety, in a Task Force Report to the Commission (NRC, 2011b). Around the same time, the NRC Office of Nuclear Regulatory Research initiated a project evaluating the consequences of a beyond-design-basis earthquake affecting an SFP at a Mark I boiling-water reactor in the United States. The results of this study, hereafter referred to as the Spent Fuel Pool Study (SFP study), were later documented in NUREG-2161 and SECY-13-0112, Enclosure 1, and are summarized in Enclosure H-5 of this appendix.

In accordance with Commission direction, the staff prioritized its recommendations in SECY-11-0137, "Prioritization of Recommended Actions to Be Taken in Response to Fukushima Lessons Learned." The staff identified expedited transfer of spent fuel to dry cask storage as an additional issue that was not identified in the Task Force Report but may warrant further consideration. In SECY-12-0025, "Proposed Orders and Requests for Information in Response to Lessons Learned from Japan's March 11, 2011, Great Tōhoku Earthquake and Tsunami," dated March 9, 2012, the staff prioritized this issue in the Tier 3 category, since it required further staff study to determine whether it warranted regulatory action. The staff also proposed two orders to the Commission that would increase SFP safety by (1) requiring installation of enhanced SFP instrumentation and (2) developing additional strategies and guidance to mitigate beyond-design-basis phenomena by maintaining or restoring SFP cooling, core cooling, and containment capabilities.

The Commission approved these orders aimed at improving spent fuel safety:

1) Order EA-12-049, "Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events," dated March 12, 2012"

   This Order requires licensees to develop, implement, and maintain guidance and strategies to maintain or restore SFP cooling capabilities, independent of alternating current power, following a beyond-design-basis external event.

2) Order EA-12-051, "Issuance of Order to Modify Licenses with Regard to Reliable Spent Fuel Pool Instrumentation," dated March 12, 2012"

   This Order requires licensees to install reliable means of remotely monitoring wide-range SFP levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event.

In SECY-12-0095, "Tier 3 Program Plans and 6-Month Status Update in Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," dated July 13, 2012, the staff outlined a five-step plan to evaluate the Tier 3 issue of whether regulatory action to expedite the transfer of spent fuel to dry cask storage was needed.

In a memorandum to the Commission entitled, "Updated Schedule and Plans for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," dated May 7, 2017, the staff provided a shortened three-phase plan for resolving the Tier 3 Issue on expedited transfer of spent fuel. The first phase of the plan was to conduct a regulatory analysis, leveraging results and insights from the ongoing SFP study, to determine whether a substantial increase in public health and safety can be achieved through an expedited transfer to dry storage casks. Then, if the results of the regulatory analysis indicated that it warranted additional study, the staff would proceed to Phases 2 and 3 of the plan and perform more detailed analyses using refined assumptions to confirm the need for regulatory action. The staff provided its findings from the Phase 1 study to the Commission in COMSECY-13-0030, which are summarized below.

Regulatory Alternatives

The staff considered two regulatory alternatives in its analysis:

- Option 1: Maintain the existing spent fuel storage requirements (regulatory baseline). This option, hereafter referred to as the regulatory baseline, refers to the case in which the Commission opts to continue with the existing licensing requirements for spent fuel storage rather than require the expedited transfer of spent fuel from SFPs to dry storage. The existing regulations require that spent fuel, which is stored in SFPs in high-density racks, be moved from SFPs into dry cask storage only when necessary to accommodate spent fuel being offloaded from the core. In addition, the SFP must always allocate enough space to accommodate at least one full core of reactor fuel in case of emergencies or other operational contingencies. The regulatory baseline assumed that all applicable requirements and guidance to date have been implemented, but it assumed no implementation for any related generic issues or other staff requirements or guidance that were unresolved or still under review. For the regulatory analysis, the baseline condition assumed that spent fuel was stored in high-density racks in a relatively full SFP, and that there was full compliance with all regulatory requirements, including those outlined in Title 10 of the *Code of Federal Regulations* (10 CFR) 50.54(hh)(2) with respect to spent fuel configuration and SFP preventive and mitigative capabilities. To increase conservatism in the analysis, for the regulatory baseline it was assumed that there was no successful mitigation of the SFP accident. In addition, because SFPs are relatively full even after using high-density storage racks, the current practice of transferring spent fuel to dry storage in accordance with 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," is assumed to continue. Lastly, although the it was assumed that licensees had implemented the requirements of Order EA-12-049 and Order EA-12-051 to enhance their ability to respond to beyond-design-basis events, the staff's evaluation did not quantitatively consider the capabilities implemented to satisfy these requirements. The regulatory baseline represents the status quo against which the second alternative is compared.

- Option 2: Expedite the transfer of spent fuel from SFPs to dry cask storage (low-density SFP). For this alternative, spent fuel assemblies that have been cooled in the SFP for at least 5 years after discharge would be expeditiously moved from the SFP to dry cask storage beginning in 2014 to achieve and maintain low-density loading of spent fuel in the existing high-density racks. For this option, the SFP would have a lower long-lived radionuclide inventory, a lower overall heat load, and a slightly higher water inventory because of the removed spent fuel assemblies. On the other hand, loading, handling, and moving casks to achieve this configuration increase the cost and risk impacts associated with this alternative. Therefore, to maximize the delta benefit of this alternative relative to the status quo (i.e., Option 1), the staff's analysis conservatively did not include these additional costs and risks associated with transferring and handling casks in their analyses. The staff also assumed that mitigative actions in accordance with 10 CFR 50.54(hh)(2) were successful to further increase the regulatory benefit of this alternative, and, similar to Option 1, did not quantitatively consider the requirements of Order EA-12-049 and Order EA-12-051 in the evaluation.

Safety Goal Evaluation

As part of its two-part regulatory analysis, the staff performed a safety goal screening evaluation to determine whether requiring the expedited transfer of spent fuel to dry cask storage would provide a significant safety benefit compared to the regulatory baseline, regardless of whether the action would be cost-beneficial. The staff performed the safety goal screening evaluation by comparing the calculated risks to the public from the severe accidents at the plants considered in this study to the two quantitative health objectives (QHOs) for average individual prompt fatalities and average individual latent cancer fatalities, as outlined in the NRC's Safety Goals Policy Statement (NRC, 1986). These QHOs, which are subsequently used to determine whether the NRC's safety goals are met, are as follows:

(1) The risk to an average individual near a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed 1/10 of 1 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.

(2) 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 1/10 of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

For an average individual within 1.6 kilometers (1 mile), the prompt fatality QHO is $5x10^{-7}$ per year as estimated in NUREG-0880, Revision 1, "Safety Goals for Nuclear Power Plant Operation," issued May 1983. The staff's analysis for expedited transfer of spent fuel showed that there are no offsite early fatalities from acute radiation effects, despite the large releases for some low-probability accident progressions analyzed.

The cancer fatality QHO listed in NUREG-0880, Revision 1, is $2x10^{-6}$ per year for an average individual living within 16 kilometers (10 miles) of a nuclear power plant site. The staff calculated an updated QHO value for comparison, using the most up-to-date estimate of the number of cancer fatalities and the total U.S. population at the time, which yielded a risk of $1.84x10^{-3}$ per year. One-tenth of 1 percent of this value results in a QHO of $1.84x10^{-6}$ per year, which is lower than the value listed in NUREG-0880.

The staff determined the risk of latent cancer fatalities to a population living near a nuclear power plant by multiplying the bounding frequency of damage to spent fuel ($3.46x10^{-5}$ per year) with the estimate from the SFP study for conditional individual latent cancer fatality risk within a 16-kilometer (10-mile) radius ($4.4x10^{-4}$ per year). This yielded a conservative high estimate of individual latent cancer fatality risk of $1.52x10^{-8}$ cancer fatalities per year for an SFP accident, which is less than one percent of the $1.84x10^{-6}$ per year QHO calculated above.

The staff noted three important limitations to the above evaluation:

(1) The safety goals outlined in the Safety Goal Policy Statement are intended to encompass all accident scenarios at a nuclear power plant site, while this analysis only considered initiating events that challenge the integrity or cooling of the SFP, which are the most important contributors to SFP risk.

(2) Although an SFP accident might affect larger areas and more people than a reactor accident, protective actions, such as relocation of the public, would result in the risks to individuals beyond 16 kilometers (10 miles) being similar to the risk to individuals located closer to the plant.

(3) The total or cumulative radiation dose to the population might be higher for an SFP accident than for a reactor accident, even though the risk to individuals living near or far from the plant remains below the QHOs.

Based on these results, the staff concluded that the continued use of high-density loadings in SFPs at nuclear power plants does not challenge the NRC's safety goals. Expediting transfer of spent fuel into dry cask storage would provide no more than a minor safety improvement.

Technical Evaluation

Description of Representative Plants

The staff organized U.S. SFPs into seven groups based on spent fuel configuration, rack designs, and SFP capacities, as shown in Table H-34.

Table H-34 SFP Groupings Used for the Staff's Technical and Cost-Benefit Analyses

SFP Group No.	Description	No. of Reactor Units	No. of SFPs	Average Year When Reactor Operating License Expires
1	Boiling-water reactors (BWRs) with Mark I and Mark II containments and with nonshared SFPs	31	31	2037
2	Pressurized-water reactors (PWRs) and BWRs with Mark III containments with nonshared SFPs	49	49	2040
3	AP1000 SFPs	4	4	2078
4	Reactor units with shared SFPs	20	10	2038
5	SFPs located below grade[1]	(these are included in group 2)		
6	Decommissioned plants with spent fuel stored in pool[2,3]	7	6	N/A
7	Decommissioned plants with fuel stored in an ISFSI using dry casks	21	N/A	N/A

[1.] Group 5 is a special set of currently operating PWRs for which damage to the pool structure would not result in a rapid loss of water inventory.
[2] The Zion 1 and 2 decommissioned reactor units share a single SFP.
[3] Group 6 includes the GE-Hitachi Morris wet independent spent fuel storage installation (ISFSI) site.

The technical evaluations discussed in this section and the cost-benefit analyses focused on Group 1 through Group 4 in Table H-34; the analyses excluded Group 5 through Group 7 for the following reasons:

- Group 5 SFPs are less susceptible to the formation of small or medium leaks because there is no open space around the pool liner and concrete structure.

- Group 6 SFPs are no longer receiving spent fuel discharged from the reactor following decommissioning, and several plants had extended plant outages before announcing cessation of plant operation.

- The spent fuel in Group 7 is already in dry cask storage.

The analyses also included operational strategies such as those used to expand onsite storage.

Spent Fuel Pool Accident Modeling

The analyses described relied heavily on the models and data used in the SFP study. NUREG-2161 and SECY-13-0112, Enclosure 1, provide more detailed information about the models developed for the SFP study. This subsection focuses on the most relevant technical information that will enable comprehension of the cost-benefit analyses described in the next section.

Seismic Hazard Model and Characterization of Seismic Event Likelihood

The analyses used the 2008 U.S. Geological Survey seismic hazard model that was available at the time (and used for the SFP study) to evaluate seismic hazards at central and eastern U.S. nuclear plants. Although this model considered hazards at western U.S. sites (e.g., Diablo Canyon), the accident analyses did not include western sites because they were not addressed in Generic Issue 199,[35] which only focused on central and eastern U.S. sites. Using peak ground acceleration and hazard exceedance frequency data from the U.S. Geological Survey, the staff determined that the hazard exceedance frequency curves of the Peach Bottom Atomic Power Station (Peach Bottom), the reference plant used for the SFP study, bound those of reactors in SFP Group 1 through Group 4 over a wide peak ground acceleration range.

To translate hazard exceedance frequencies into seismic initiating event frequencies, the staff also partitioned the peak ground acceleration ranges for Peach Bottom and for sites in SFP Group 1 through Group 4 into four discrete bins. Since the SFP study demonstrated that damage to the SFP and other related structures was not credible for seismic bins 1 and 2, the staff only used seismic initiator event frequencies from bins 3 and 4 of each SFP group (and Peach Bottom). Specifically, the analyses used seismic initiating event frequencies from bins 3 ($1.7\text{x}10^{-5}$ per year) and 4 ($4.9\text{x}10^{-6}$ per year) for Peach Bottom for both the low- and base-case analyses because these hazard exceedance frequencies bound most of the other reactor sites. To account for some reactor site hazard exceedance frequencies exceeded those of Peach Bottom for bins 3 and 4, for each SFP group, the analyses used the site with the largest plant exceedance frequencies in bins 3 and 4 to generate high-estimate seismic initiating event frequencies for subsequent sensitivity analyses (see Table H-35).

Consequence Analyses

The MELCOR Accident Consequence Code System (MACCS[36]) code was used to model atmospheric transport and dispersion, emergency response, and long-term consequences. The atmospheric transport and dispersion model used for these analyses was based on the Peach Bottom MACCS results described in the SFP study. The MACCS model for Peach Bottom used a straight-line Gaussian plume segment model. For both the SFP study and this study, the atmospheric release of radionuclides was discretized into up to 1-hour plume segments to account for variations in the release rate and the changes in wind direction. Meteorological data used for the MACCS analyses consisted of 1 year of hourly meteorological data (i.e., 8,760 data points for each meteorological parameter) for Peach Bottom evaluated in the SFP study. The specific year of meteorological data chosen for Peach Bottom was 2006, and stability class data were derived from temperature measurements at two elevations on the site meteorological towers.

The study used population densities and site distribution characteristics for SFPs in the United States to generate the site population and economic data required for MACCS and cost-benefit analyses. The SFP sites were binned based on average population densities within 80 kilometers (50 miles) of the sites, and representative sites were selected to represent various population densities. Peach Bottom, Surry Power Station, Palisades Nuclear Plant, and Point Beach Nuclear Plant represented population densities in the 90th percentile, the mean, the

35 https://www.nrc.gov/about-nrc/regulatory/gen-issues/dashboard.html#genericIssue/genericIssueDetails/3

36 At the time of this analysis, the MACCS code was called the "MACCS2" code, a leftover notation from the time that the original MACCS code was substantially upgraded to Version 2. Since then, the staff has referred to the code as the "MACCS" code and notes the version number of the code used in a particular analysis since code development and maintenance continues.

median, and the 20th percentiles, respectively. For each representative site, site population and economic data were created for 16 compass sectors and interpolated onto a 64-compass sector grid for better spatial resolution for consequence analyses. The staff escalated 2000 census data and 2002 economic data to 2011 values.

Population densities and distributions near SFP locations representing the 90th, mean, median, and 20th percentiles were used for respective high-, base-, median-, and low-estimate sensitivity studies of site population demographics. The study used these data as additional inputs into MACCS calculations to assess the effect of population density on the averted public health (accident) attribute. Since an SFP fire could affect public health consequences beyond 80 kilometers (50 miles), sensitivity analyses were also conducted using base-case assumptions and the standard value ($2,000 per person-rem), along with a sensitivity value ($4,000 per person-rem) for the person-rem conversion factor. The study used the $4,000 per person-rem sensitivity value because the staff was reassessing the dollar per person-rem factor at the time as part of its efforts to update NUREG-1530, "Reassessment of NRC's Dollar Per Person-Rem Conversion Factor Policy," issued December 1995, and Revision 1, issued August 2015 (NRC, 1995b; NRC, 2017b).

The study evaluated the relationship between population densities, distribution characteristics, and offsite property values near SFP sites by conducting sensitivity analyses in which the site population densities and distributions were varied. The site populations, distributions, and economic data for the high-, base-, median-, and low-estimate cases described above served as additional input into the MACCS calculations that otherwise used values specific to the reference plant. The staff also evaluated the impact on offsite property costs as a result of extending offsite consequences beyond 80 kilometers (50 miles). In this case, the base-case assumptions and the intermediate protective action guidelines criterion were used, as explained below.

The SFP study used the emergency response model in MACCS to model doses, health effects, and emergency response during the 7-day period following the start of a release during a severe accident. The long-term phase, which is the period following the 7-day emergency phase, was modeled for 50 years to calculate consequences from exposure of an average person. The habitability criterion used in MACCS, to determine whether land is inhabitable after decontamination, was 2 rem in the first year and 500 millirem (mrem) each year thereafter for the base-case evaluations. This criterion was based on the U.S. Environmental Protection Agency's protective action guidelines as outlined in EPA-400/R-17/001, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," issued January 2017 (EPA, 2017). However, for habitability, some States (e.g., Pennsylvania) have adopted a habitability criterion of 500 mrem annually. To account for the uncertainties in the way in which States define their habitability criteria, the staff also performed sensitivity studies in which the low estimate case used 500 mrem per year, while the high-estimate case used a conservative 2 rem per year.

Cost-Benefit Analysis

A cost-benefit analysis informed the Commission's decision whether to expedite spent fuel transfer to dry cask storage. This analysis was more expansive than that performed for the SFP study, as it evaluated SFP configurations at all U.S. nuclear power plants and it incorporated insights from the SFP study and other previous studies, where possible.

Methodology

The staff first identified the attributes that would be impacted by expedited fuel transfer and performed quantitative and qualitative analyses on those attributes, including public health (accident) and occupational health (routine and accident), onsite property, offsite property, industry implementation and operational activities, and NRC implementation and operational activities. The analysis did not include the NRC's implementation and operational activity costs; this simplification is acceptable because it is consistent with the approach to maximize the benefit of the alternative.

The staff determined the costs and benefits associated with each attribute for each alternative, converting them into monetary values where practicable and discounting them to a net present value. Specifically, the staff used a constant 7 percent discount rate as a base-case value and used 3 percent as a sensitivity value to approximate the real rate of return on long-term government debt, which is a proxy for the real rate of return on savings. In addition, the Office of Management and Budget (OMB) Circular No. A-4, "Regulatory Analysis," dated September 17, 2003, suggests using a lower but positive discount rate, in addition to the discount rates of 3 percent and 7 percent, if the decisionmaking will have important intergenerational benefits. Therefore, for this study, the staff included a 2 percent discount rate to represent the lower bound for the certainty-equivalency rate in 100 years. The staff analyzed the total discounted quantitative costs and benefits for each alternative to determine whether there was a positive benefit for expedited transfer. The staff also considered qualitative costs and benefits in assessing whether there was a positive benefit.

The staff performed a sensitivity analysis to identify key input parameters that have the greatest impact on the results. Starting with the parameters for the base case, it varied the input parameters to generate low- and high-estimates that it compared with the base-case results to determine the sensitivity of the results to the input parameter. The results of these analyses indicated that, in addition to discount values used for present value calculations, dollar per person-rem conversion factors, calculated consequences from the site, habitability criteria, and seismic initiator frequency were also key input parameters that strongly affected the net results. Table H-35 summarizes the base-case and sensitivity values used for the key input parameters.

Table H-35 Key Input Parameters Used for Sensitivity Analyses

Input Parameter	Methodology	
	Base Case Value	Sensitivity Value(s)
Net Present Value (NPV)	7% NPV	2 and 3% NPV
Dollar per person-rem Conversion Factor	$2,000	$4,000
Calculated Consequences from Site	50 miles	Beyond 50 miles
Habitability Criteria	2 rem in the first year and 500 mrem each year thereafter	500 mrem per year and 2 rem per year
Seismic Initiator Frequency[a]	Bin 3: $1.65x10^{-5}$ per year Bin 4: $4.90x10^{-6}$ per year	Bin 3: $2.24x10^{-5}$–$5.64x10^{-5}$ per year Bin 4: $7.09x10^{-6}$–$2.00x10^{-5}$ per year

[a] As discussed in the SFP study, damage to the SFP and other relevant structures, systems, and components is not credible for events in bins 1 and 2.

The staff made its recommendation on the implementation of each alternative based on qualitative attributes, uncertainties, sensitivities, and the quantified costs and benefits taken from quantitative attributes. If the quantified and qualified benefits were greater than the quantified and qualified costs, then the staff recommended the alternative be implemented. Otherwise, the staff recommended that the alternative not be implemented.

Cost-Benefit Analysis Results

Table H-36 summarizes the net benefits (i.e., the sum of total benefits and total costs) for each SFP group. The table includes the corresponding values obtained from additional sensitivity analyses in which the discount rate of 7 percent, which the NRC uses for regulatory decisionmaking, was varied to 2 percent and 3 percent in accordance with the recommendations in OMB Circular A-4. In addition to the conservative assumptions used to generate the base-case values, low- and high-estimates are provided that combine the range of expected SFP attributes to model the range of pool accidents postulated.

Table H-36 Summary of Net Benefits for Each Spent Fuel Pool Group*

SFP Group No.	Low Estimate (2012 million dollars)			Base Case (2012 million dollars)	High Estimate (2012 million dollars)		
	2% NPV	3% NPV	7% NPV	7% NPV	2% NPV	3% NPV	7% NPV
1	($53)**	($55)	($52)	($45)	$70	$54	$21
2	($51)	($54)	($51)	($45)	$86	$67	$26
3	($42)	($36)	($17)	($12)	$66	$45	$17
4	($49)	($50)	($49)	($39)	$160	$130	$74

* Note: The values listed in COMSECY-13-0030, Enclosure 1, have been rounded to two significant figures here.
** Negative values are shown using parentheses (e.g., negative $53 is displayed as ($53)).

Attributes that led to net costs for SFP Group 1 through Group 4 are industry implementation and occupational health (routine) costs, with implementation costs far surpassing routine occupational health costs. For Group 1, Group 2, and Group 4, these costs are dominated by the additional capital costs for the dry storage containers (DSCs) and loading costs for the storage systems to achieve low-density storage in the SFP above that required for the regulatory baseline. Since the spent fuel stored in Group 3 SFPs is not expected to require dry storage until 2038, additional costs beyond the DSC capital costs and loading costs include ISFSI annual operation and maintenance costs required to establish the ISFSI and store spent fuel there 15 years earlier than in the regulatory baseline.

Positive attributes (i.e., benefits and cost offsets) that offset the net costs described above are public health (accident), occupational health (accident), offsite property, and onsite property. For all groups, the offsite property cost offset is the largest contributor to the benefits, the majority of which occur during the long-term phase. However, as Table H-37 illustrates, these benefits and cost offsets do not create a positive net benefit for low-, high-, or base-case-estimates with any of the discount rates applied.

The staff performed sensitivity analyses to provide additional consideration for the safety goal screening evaluation. Table H-37 summarizes the results of the sensitivity analyses considering the combined effects of adjusting the dollar per person-rem conversion factor from $2,000 to $4,000 and of extending consequence analyses beyond 80 kilometers (50 miles) from the site.

Table H-37 Net Benefits for Low-Density SFP Storage for Groups 1–4 from Combined Sensitivity Analyses that Analyzed Consequences Beyond 80 kilometers (50 Miles) and Using an Adjusted Dollar per Person-Rem Conversion Factor

SFP Group No.	Low Estimate (2012 million dollars)*			Base Case (2012 million dollars)*			High Estimate (2012 million dollars)*		
	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV	2% NPV	3% NPV	7% NPV
1	($51)**	($54)	($51)	$9.5	$0.17	($15)	$880	$779	$506
2	($48)	($51)	($49)	$19	$7.7	($12)	$1,100	$916	$569
3	($39)	($33)	($16)	$32	$21	$6.8	$749	$563	$233
4	($45)	($47)	($44)	$40	$28	$5.8	$1,900	$1,600	$1,100

* Note: the original values for this analysis listed in COMSECY-13-0030, Enclosure 1, have been rounded to two significant figures.
** Negative values are shown using parentheses (e.g., negative $51 is displayed as ($51)).

The sensitivity results provided in Table H-37 show that there are cases using conservative assumptions for each SFP group in which the low-density spent fuel storage alternative was cost-justified. However, after considering the analysis results, operating history, and limited safety benefits of possible plant changes, the staff concluded that further study would be unlikely to support future actions requiring expedited transfer.

Summary and Conclusion

The staff performed a regulatory analysis that included all U.S. SFPs to determine whether expedited transfer of spent fuel from SFPs to dry cask storage was warranted. As part of the regulatory analysis, the staff conducted a technical evaluation using insights from recently completed SFPs, a safety goal screening evaluation, and a cost-benefit analysis. The results of the technical evaluation of the consequences of seismic events impacting four different categories of SFPs indicated that no offsite fatalities were expected to occur, similar to the results obtained from the SFP study and other studies, and that the predicted long-term exposure of the population, which could result in latent cancer fatalities, was low.

The safety goal screening evaluation revealed that SFP accidents are a small contributor to the overall risks for public health and safety (less than 1 percent of the QHOs), and therefore any reductions in risk associated with expedited transfer of spent fuel only would have a marginal safety benefit. In addition, the cost-benefit analysis demonstrated that the added costs of expediting transfer of spent fuel to dry cask storage were not warranted considering the marginal safety benefits that would result. As part of the analysis, the staff identified attributes affected by expedited transfer and analyzed them quantitatively and qualitatively, where possible. When considering the discount rates combined with very conservative SFP assumptions, the costs of implementing expedited transfer greatly outweighed the benefits of doing so. However, the combination of high estimates for important parameters used in subsequent sensitivity analyses resulted in large economic consequences, such that the calculated benefits from expedited transfer of spent fuel to dry cask storage for those cases outweighed the associated costs. For those cases, the staff concluded that there was only a marginal safety improvement in terms of public health and safety, asserting that the assumptions made in the analyses were selected in a generally conservative manner such that the base case is the primary basis for the staff’s recommendation.

Based on the analyses presented in COMSECY-13-0030, the staff concluded that additional studies were not needed to reasonably conclude that the expedited transfer of spent fuel to dry cask storage would provide only a marginal increase in the overall protection of public health and safety. The staff also informed the Commission that it recommended no further regulatory action for the resolution of this Tier 3 issue.

Staff Non-concurrence

In accordance with Management Directive 10.158, "NRC Non-Concurrence Process," dated March 14, 2014, a member of the NRC technical staff submitted a non-concurrence on COMSECY-13-0030. Enclosure 2 to COMSECY-13-0030 provides documentation associated with this non-concurrence.

The non-concurrence raised several issues with the detailed analyses performed in support of COMSECY-13-0030, including (1) other potentially cost-beneficial approaches to improving the safety of SFPs should have been evaluated, in addition to Option 2, (2) the base case analysis should have used different assumptions for factors that were ultimately evaluated only as sensitivity analyses (e.g., the dollar per person-rem conversation factor, the region over which offsite radiological consequences are aggregated), (3) the staff should acknowledge the limitations of using safety goals and QHOs that were developed for reactor accidents to determine whether a proposed regulatory action pertaining to SFP safety would constitute a substantial safety enhancement, and (4) the presentation of results should have provided a more balanced and neutral view of the range of findings that were obtained by using the high-estimate cases and sensitivity analyses.

The staff made several improvements to COMSECY-13-0030 in response to the concerns raised in the non-concurrence. However, after considering the analysis results, operating history, and limited safety benefits of possible plant changes, the staff ultimately concluded that additional studies would be unlikely to support a requirement to expedite transfer of spent fuel from SFP storage to dry cask storage to achieve a low-density SFP loading configuration.

Commission's Response to the Staff's Analysis and Recommendations

In the staff requirements memorandum for Staff Requirements Memoranda (SRM)-COMSECY-13-0030, dated May 23, 2014, "Staff Evaluation and Recommendation for Japan Lessons-Learned Tier 3 Issue on Expedited Transfer of Spent Fuel," the Commission approved the staff's recommendation that the Tier 3 Japan lessons-learned activities for expedited transfer be closed, and that no further generic assessments be conducted. The Commission also directed the staff to perform several other related activities for completeness and closure of the Tier 3 issue, including modifying the regulatory analysis provided in COMSECY-13-0030 to explain why the 1x8 configuration would not provide a substantial increase in safety. The staff addressed the above issues in SECY-15-0059, "Seventh 6-Month Status Update on Response to Lessons Learned from Japan's March 11, 2011, Great Tohoku Earthquake and Subsequent Tsunami," Enclosure 3, dated April 9, 2015 (NRC, 2015e).