June 10, 2022 Docket No. 99902043

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Topical Report "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772, Revision 3

REFERENCES: 1. NuScale Letter to NRC, "NuScale Power, LLC Submittal of 'Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones,' Revision 2, TR-0915- 17772," dated August 4, 2020 (ML20217L423)

NuScale Power, LLC (NuScale) hereby submits Revision 3 of the "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," (TR-0915-17772). The purpose of this submittal is to request that the NRC review and approve the use of the methodology for establishing the technical basis for plume exposure emergency planning zones as described in the topical report.

Enclosure 1 contains the proprietary version of the report entitled "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," TR-0915-17772, Revision 3. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of the report.

This letter makes no regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions, please contact Liz English at 541-452-7333 or at EEnglish@nuscalepower.com.

Mark W. Shaver

Sincerely,

Mark W. Shaver
Manager, Licensing
NuScale Power, LLC

Distribution: Michael Dudek, NRC
Getachew Tesfaye, NRC
Bruce Bavol, NRC
Alina Schiller, NRC

Enclosure 1: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772-P, Revision 3, proprietary version

Enclosure 2: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772, Revision 3, nonproprietary version

Enclosure 3: Affidavit of Mark Shaver, AF-119991

Enclosure 1:

"Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772-P, Revision 3, proprietary version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

Enclosure 2:

“Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones,” TR-0915-17772, Revision 3, nonproprietary version

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330 Office 541.360-0500 Fax 541.207.3928
www.nuscalepower.com

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Department of Energy Acknowledgement and Disclaimer

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Licensing Topical Report

Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites

June 2022

Revision 3

Docket: PROJ99902043

NuScale Power, LLC
1100 NR Circle Blvd., Suite 200
Corvallis, Oregon 97330
www.nuscalepower.com

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List of Affected Pages

Revision Number	Page Number	Explanation
2	All pages	Revised to address public and NRC comments. Major modifications to single module screening criteria and defense-in-depth assessment. Other minor methodology changes and editorial revisions.
3	Most pages	Major revision to address RAI 9828.

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Abstract

The purpose of this licensing topical report (LTR) is to provide the technical basis for NuScale's plume exposure pathway emergency planning zone (EPZ) sizing methodology. The ingestion EPZ is not addressed in this methodology, as the determination of this distance is dependent on land usage that is site-specific. The methodology is informed by the Nuclear Energy Institute (NEI) risk-informed EPZ methodology (Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone, Reference 6.1.5) and extends this risk-informed methodology to address the issue of determining the appropriate accident sequences to be included in the EPZ technical basis, and to consider a consequence orientation in the approach. The screening of accident sequences includes the use of quantitative insights from probabilistic risk assessment (PRA) as well as application of engineering insights emphasizing safety margin and layers of defense-in-depth. The screening methodology includes consideration of all hazards and operating modes and also contains integrated assessment of both multi-module effects, when applicable, and uncertainty analysis. Based on the accident sequence screening, the risk results, including source terms and off-site dose versus distance, will serve as the basis for a plume exposure EPZ size. The methodology is intended for use by NuScale light water small modular reactors (SMRs).

The LTR contains the plume exposure EPZ size methodology for which NRC approval is being sought. In some instances, NuScale design certification application information is used as examples to explain the methodology; however NuScale design information does not form the basis of the methodology.

The topical report requests an NRC review of NuScale's plume exposure EPZ sizing methodology. NuScale also requests, as part of this review and associated comment resolution, that the NRC provide a safety evaluation report on the design-specific sizing methodology, including the following:

1. A conclusion that the NuScale-proposed plume exposure EPZ methodology in the LTR, when supported by design-specific information and appropriately implemented by the applicant, is an acceptable approach for justifying the plume exposure EPZ size.
2. Identification of issues related to the EPZ technical basis that are to be resolved prior to or as part of the application review process.

To aid in the NRC's review, each section of the topical report individually identifies the approval request and associated acceptance criteria.

Executive Summary

The purpose of this LTR is to provide a methodology to establish the technical basis for plume exposure EPZ sizing. The methodology is intended for use by NuScale light water small modular reactors (SMRs). Nuclear power plant emergency planning regulatory requirements are codified under Emergency Plans, 10 CFR 50, Part 50.47 (Reference 6.1.1), and Emergency Planning and Preparedness for Production and Utilization Facilities, 10 CFR Part 50 Appendix E (Reference 6.1.2). The responsibility for reviewing emergency planning lies with the U.S. Nuclear Regulatory Commission (NRC) in coordination with the Federal Emergency Management Agency (FEMA). The current regulatory plume exposure EPZ for power reactors is 10 miles, but there is a provision for a different EPZ size for reactors with a thermal power of 250 MWt or less on a case-by-case basis. This report describes a methodology to establish the technical basis for plume exposure EPZ sizing. The ingestion EPZ is not addressed in this methodology, as the determination of this distance is dependent on land usage that is site-specific.

NuScale requests, as part of the review and associated comment resolution of this LTR, that the NRC provide a safety evaluation report on the plume exposure EPZ sizing methodology. The methodology herein, when supported by design-specific information and appropriately implemented by the applicant, is an acceptable approach to plume exposure EPZ sizing. To aid in the NRC's review, each section of the topical report individually identifies the approval request and associated acceptance criteria.

The methodology described in this report is informed by the 2013 NEI White Paper framework and incorporates concepts from the original, generic 1978 EPZ size basis (Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans, NUREG-0396, Reference 6.1.3) in that the objective goal is dose-based linked to considerations of consequences. However, the methodology is applied utilizing PRA information supported by a comprehensive evaluation of severe accident sequences. It includes consideration of internal events, external hazards, and all modes of operation, as well as other PRA risks.

The main body of the LTR, in Section 3.0, presents the EPZ size methodology for which NRC approval is sought. In some instances, NuScale design certification application information is used as examples to explain the methodology; however NuScale design information does not form the basis of the methodology. The methodology requires compilation of accident sequences from the PRA for all initiators and screens the sequences for inclusion in the EPZ technical basis based upon multiple criteria.

The methodology first determines the spectrum of accident sequences to be evaluated for EPZ. The screening of accident sequences includes the use of quantitative insights from the PRA, including consideration of uncertainty, as well as application of engineering insights emphasizing safety margin and layers of defense-in-depth. Both "less severe" and "more severe" sequences are evaluated, differentiated by containment status (intact or failed). Based on the accident sequence screening, the risk results, including source terms and off-site dose versus distance, serve as the basis for a plume exposure EPZ size methodology.

The dose criteria for the NuScale EPZ methodology have been defined based on the original EPZ bases, as noted, and the Environmental Protection Agency's (EPA's) protective action guides (PAGs) (PAG Manual, EPA-400/R-17/001, Reference 6.1.4), applied to the sequences as

follows: (1) 1 to 5 rem total dose effective equivalent (TEDE) for a design-basis accident (DBA); (2) 1 to 5 rem TEDE for less severe sequences; and (3) 200 rem whole body acute dose for more severe sequences.

Using the risk-informed methodology developed to select appropriate accident sequences, NuScale has also developed a method to evaluate the source term and dose consequence for both less severe and more severe accidents, as presented in Section 4.0. This methodology includes integrated uncertainty analysis.

Finally, the EPZ methodology addresses multi-module accidents. The multi-module accident methodology focuses on multi-module risks associated with shared initiating events and structures, as well as shared systems among modules. The multi-module methodology is applicable to designs with multiple co-located modules.

In summary, the NuScale methodology for establishing the technical basis for plume exposure EPZ sizing considers source terms and dose consequences. The methodology, when implemented with design information as part of an application, provides a basis for sizing the plume exposure EPZ. The methodology is applicable to any EPZ size, including the site boundary. The final EPZ size that results from applying the methodology is the smallest distance at which the dose consequences of screened-in accident sequences meet their respective dose criterion. Based on the results of applying the methodology, the final EPZ size may differ from the current 10 mile requirement.

1.0 Introduction

1.1 Purpose

The purpose of this licensing topical report (LTR) is to provide a methodology and criteria that can be implemented to establish the design-specific and site-specific plume exposure emergency planning zone (EPZ) size. The methodology is intended for use by NuScale light water small modular reactors (SMRs). The purpose of submitting this LTR is to provide information to the U.S. Nuclear Regulatory Commission (NRC) to facilitate efficient and timely review of the NuScale plume exposure EPZ sizing methodology. NuScale also requests, as part of this review and associated comment resolution, that the NRC provide a safety evaluation report on the plume exposure EPZ sizing methodology.

1.2 Scope

This report provides a methodology for determining an appropriate plume exposure EPZ. The ingestion EPZ is not addressed in this methodology, as the determination of that distance is dependent on land usage, which is site-specific. The NuScale methodology expands on the Nuclear Energy Institute (NEI) risk-informed EPZ methodology (Reference 6.1.5).

This report is based on the following regulatory guidance and technical considerations:

- methodology designed to be structured and repeatable
- NRC EPZ documents (NUREG-0396 [Reference 6.1.3], Generalized Dose Assessment Methodology for Informing Emergency Planning Zone Size Determinations [Reference 6.1.11], Required Analyses for Informing Emergency Planning Zone Size Determinations [Reference 6.1.12], Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors, SECY-97-020 [Reference 6.1.8], Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors, SECY-11-0152 [Reference 6.1.6], and Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies, SECY-15-0077 [Reference 6.1.7])
- risk-informed methods to determine the spectrum of accident sequences to be evaluated, including multi-module events and external events
- analysis of uncertainties

The main body of the LTR contains the plume exposure EPZ size methodology for which NRC approval is sought. In some instances, NuScale design certification application (DCA) information is used as examples to explain the methodology; however NuScale design information does not form the basis of the methodology. This LTR is not part of the NuScale DCA.

The use of this methodology to determine final EPZ size will occur when an application is submitted to the NRC to construct and operate a NuScale light water SMR design. The most likely mechanism is a combined license (COL) application; however it is acknowledged that other regulatory processes exist. For simplicity, "COL applicant" and

"COL application" are used throughout this LTR to refer to implementation of the methodology.

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
AEF	annual exceedance frequency
ATD	atmospheric transport and dispersion
ATWS	anticipated transient without scram
CDF	core damage frequency
COL	combined license
DBA	design-basis accident
DBST	design-basis source term
DCA	design certification application
DCF	dose conversion factor
DOE	Department of Energy
EDMG	extensive damage mitigating guideline
EOP	emergency operating procedure
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ESP	early site permit
FEMA	Federal Emergency Management Agency
FSAR	final safety analysis report
INSAG	International Nuclear Safety Advisory Group
LOLA	loss of large areas
LTR	licensing topical report
LWR	light water reactor
MACCS	MELCOR Accident Consequence Code System
NEI	Nuclear Energy Institute
NPM	NuScale Power Module
NRC	U.S. Nuclear Regulatory Commission
NuScale	NuScale Power, LLC
PAG	protective action guide
PRA	probabilistic risk assessment
RG	regulatory guide
SAMG	severe accident management guideline
SFP	spent fuel pool
SMR	small modular reactor
SOARCA	state-of-the-art reactor consequence analyses
SRM	staff requirements memorandum
SSC	structure, system, and component
TEDE	total effective dose equivalent

Table 1-1 Abbreviations (Continued)

Term	Definition
TVA	Tennessee Valley Authority
UHS	ultimate heat sink

Table 1-2 Definitions

Term	Definition
acute whole body dose	Dose produced by short duration exposure, usually over a matter of minutes, to deeply penetrating radiation (i.e., x-rays, gammas, or neutrons able to reach internal organs) uniformly over a significant portion of the body. Acute whole body dose is represented by dose to red bone marrow and is used to predict the onset of acute radiation syndrome deterministic health effects.
beyond-design-basis accidents	Events whose assumptions for failures or initiating events are outside of the plant design basis.
conditional probability	In PRA, a conditional probability can be calculated for containment failure, core damage, or large release given the knowledge that a prior event has occurred.
core damage	Uncovery and heatup of the reactor core to the point at which prolonged oxidation and severe fuel damage are anticipated and involving enough of the core, if released, to result in off-site public health effects.
core damage frequency	Expected number of core damage events per unit of time.
defense-in-depth	An approach to designing and operating nuclear facilities that prevents and mitigates accidents that release radiation or hazardous materials. The key is creating multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth includes the use of access controls, physical barriers, redundant and diverse key safety functions, and emergency response measures.
design-basis accidents	Event sequences deterministically selected for the purpose of performing conservative deterministic safety analyses to demonstrate that design-basis accident dose requirements can be achieved by assuming that only safety-related structures, systems, and components perform as required.
design-basis source term	Postulated event with radionuclides released into an intact containment to enable deterministic evaluation of the response of a facility's engineered safety features.
emergency planning zone	An area surrounding a plant with a well-defined boundary for which emergency planning is provided, including provisions for protective actions such as evacuation and sheltering.
engineered safety feature	A structure, system, or component that is relied upon during, or following design-basis events to ensure the capability to prevent or mitigate the consequences of those events that could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) (Reference 6.1.9) excluding reactor coolant pressure boundary and reactor protection system items.
exceedance frequency	Annual rate at which a parameter exceeds a given value. As an example, for seismic hazards, the parameter used is the peak ground acceleration.
external hazard	A hazard originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release.
large release frequency	The frequency of an unmitigated release of airborne fission products from the containment to the environment such that there is a potential for significant radiological doses to the public.

Table 1-2 Definitions (Continued)

Term	Definition
plume exposure pathway EPZ	For nuclear power reactors the plume exposure pathway EPZ is an area of about 10 miles (16 km) in radius. The principal exposure sources from this pathway are- (a) whole body external exposure to gamma radiation from the plume and from deposited material; and (b) inhalation exposure from the passing radioactive plume. The time of potential exposure could range from hours to days. Current NRC regulations allow for different areas for reactors with a core power of no more than 250 MWt.
probabilistic risk assessment	A qualitative and quantitative assessment of the risk associated with plant design, operation, and maintenance that are measured in terms of frequency of occurrence of risk metrics, such as the core damage or a radioactive material release and its effects on the health of the public.
risk-based	A characteristic of decision-making in which a decision is solely based on the numerical results of a risk assessment.
risk-informed	A characteristic of decision-making in which risk results or insights are used together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to public health and safety.
Seismic Category I	Structures, systems, and components that are designed to remain functional if a safe-shutdown earthquake occurs.
sequence	A series of events (e.g., event tree sequence, accident sequence) referring to a specific event tree pathway in a PRA model that begins with an initiating event and describes the successful and unsuccessful responses of structures, systems, and components in response to the initiating event and ends in a distinct end state. Refer to Section 3.4.1 for an expanded definition of sequence in the context of sequence screening.
severe accidents	An accident event that involves extensive core damage and fission product release into the reactor vessel and containment, with potential release to the environment.
total effective dose equivalent	The sum of doses to organs or tissues that accounts for: the radiation energy absorbed by each organ or tissue through direct exposure and through a 50-year commitment period following the intake of radioactive material; the effectiveness of a type of radiation to cause biological damage to that organ or tissue; and the sensitivity of that organ or tissue to the induction of stochastic health effects.
ultimate heat sink	A set of safety-related pools of borated water that consists of the combined water volume of the reactor pool, refueling pool, and spent fuel pool. The ultimate heat sink pools are located below grade in the reactor building.

2.0 Background

The purpose of this section is to provide background on the technical and regulatory basis of the 10-mile plume exposure EPZ for the large operating plants, discuss at a high level the reasons for reconsidering EPZ size, and discuss recent industry and NRC documents that address reevaluation of EPZ size and planning elements for SMRs, including rulemaking on SMR EPZ.

Protective action zones around commercial nuclear power plants have been an NRC requirement since the early 1960s. Reactor Site Criteria, 10 CFR Part 100 (Reference 6.2.4) required that every site must have an exclusion area and a low population zone.

In 1978, the NUREG-0396 study (Reference 6.1.3), which was based on NUREG-75/014, Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants (WASH-1400), (Reference 6.2.5), provided a technical basis for a plume exposure pathway EPZ of about 10 miles (16 kilometers) and an ingestion exposure pathway EPZ of about 50 miles (80 kilometers). EPZs for the large operating plants were established by rulemaking as discussed in Section 2.1 at 10 and 50 miles to provide dose savings to the population in areas where the projected dose from design-basis accidents (DBAs) could be expected to exceed the applicable Environmental Protection Agency (EPA) protective action guides (PAG) of 1 and 5 rem total effective dose equivalent (TEDE) and severe accidents could be expected to exceed the early injury threshold of 200 rem whole body acute dose.

Several reasons to reconsider the 10-mile plume exposure EPZ for SMRs are summarized in the NEI white paper:

> "An SMR replacing an existing fossil plant, co-located at a site with industrial customers presents a unique situation. For SMRs the benefits of appropriate EPZ sizing are significant. SMRs hold significant promise in meeting energy needs worldwide for: inherently safe, scalable, economical electric power generation; electric power generation at a distance from large grid systems; and applications in addition to electric power generation such as water desalination and process heat. Successful development and deployment of these new technologies requires commensurate and timely regulatory evolution, including in the area of emergency planning (EP).
>
> There are several reasons for reconsidering EPZ sizing for SMRs. First, the SMR designs are different from traditional, large light water reactor (LWR) plants in ways which significantly reduce the potential for off-site fission product release and dose consequences (e.g., smaller core fission product inventories, improved design features, and slower accident sequence evolution). The EPZ size for SMRs should reflect their design, source terms, and severe accident dose characteristics. Second, there have been significant advancements over the last several decades in the understanding of severe accidents, fission product release and transport phenomena, consequence analysis, and effectiveness of off-site protective actions, all of which suggests smaller, slower fission product releases during accidents and reduced

> health and safety risks to the public as compared with earlier conservative analyses. Third, is that implementation of appropriate EPZ sizing can simplify interfaces between the plant operator, the surrounding communities, and any co-located customers. This benefits both the communities and the licensee, and will significantly contribute to successful deployment of SMRs in the U.S."

The concept of an EPZ size commensurate with off-site radiological risk is not new to the NRC. The staff reviewed and approved EPZ size-related exemption requests from certain reactor licensees that have since ceased operations (recent examples include Request by Dominion Energy Kewaunee, Inc. for Exemptions from Certain Emergency Planning Requirements, SECY-14-0066, and Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements, SECY-14-0118) (Reference 6.2.6 and Reference 6.2.7, respectively). The staff reviewed these exemption requests with respect to the requirements in 10 CFR 50.47, (Reference 6.1.1); 10 CFR 50, Appendix E (Reference 6.1.2); and Emergency Plans, 10 CFR 72.32 (Reference 6.2.8).

Industry believes that siting and building advanced reactors, such as SMRs, with appropriate EPZ size and planning elements will have benefits for all stakeholders. This is based on the expectation that the overall safety case and defense-in-depth, including design, operation, security, and appropriate EPZ and planning elements, will further enhance the design and safety margins and further reduce accident risk to the public.

Most of the fundamental factors providing the technical basis for emergency plan requirements and EPZ size for the current fleet of nuclear plants are similar from plant to plant; for example, core fission product inventories, reactor containment design parameters, use of active safety systems, dependence on electric power and operator actions in accident situations, and the potential for relatively fast time to core uncovery in the low likelihood event of a beyond-design-basis severe accident. Given these similarities, all operating plants in the United States meet the same emergency plan requirements, including a 10-mile plume exposure EPZ.

By contrast, the NuScale design differs significantly from large LWRs. For example, a NuScale Power Module (NPM) is a small fraction of the thermal power of existing large plants, which translates into a much lower fission product inventory. In addition, the NPM design has passive safety systems with no dependence on electric power and does not rely on operator actions to mitigate the effects of a design-basis accident for the first 72 hours following the event. Further, the passive processes that drive the transfer of heat from the reactor core to the ultimate heat sink (UHS) (namely, natural circulation, convection, and conduction) serve as an efficient and continuous means of removing decay heat during plant transients including both design-basis and beyond-design-basis events.

These differences between the NuScale design and the large operating plants, as well as the significantly reduced frequency of an accident that results in core damage, support a reduced EPZ size while providing, in the very unlikely event of core damage, the same or increased public protection as the EPZ size of the existing fleet.

The advanced designs such as NuScale offer unique opportunities to optimize emergency planning size and requirements. This optimization supports a smaller plume exposure EPZ size and appropriate, associated revisions to emergency plan requirements in 10 CFR 50.47 and 10 CFR 50, Appendix E.

2.1 Evolution of EPZ-Related Regulatory Requirements and Guidance

In 1978, the NRC issued the NUREG-0396 study, which provided a technical basis for development of emergency response plans and for EPZ size. In 1979, the NRC issued a policy statement describing the two EPZs: a plume exposure EPZ of about 10 miles and an ingestion pathway EPZ of about 50 miles. The plume EPZ is for detailed planning and rapid response, and provides a base for expansion beyond the EPZ boundary if necessary. The ingestion EPZ is for longer term actions.

Following the Three Mile Island accident, the two EPZs were included in a 1980 rulemaking establishing specific requirements for emergency plans at commercial nuclear plants. These requirements are codified in 10 CFR 50.47 and 10 CFR Part 50, Appendix E. In 10 CFR 50.47(a)(2), the NRC's determination of acceptability is tied directly to the review of the off-site plan by the Federal Emergency Management Agency (FEMA) and resulting findings. FEMA and NRC acceptance of the emergency plan is a prerequisite for approval of a COL under 10 CFR 52, Subpart C - Combined Licensees (Reference 6.2.1). The NRC approval of an early site permit (ESP) under Early Site Permits, 10 CFR 52, Subpart A (Reference 6.2.2) requires either: a no significant impediments for emergency plans assertion; or a major features emergency plan. Both 10 CFR 50.47 and 10 CFR 50 Appendix E require a 10-mile plume exposure EPZ for power reactors, but also provide for a different EPZ size for reactors with a thermal power of less than 250 MWt on a case-by-case basis. In 1980, the NRC and FEMA published a regulatory guidance document, NUREG-0654 (Reference 6.2.9), which contains criteria for preparation and evaluation of emergency response plans.

More recently, the staff has provided EPZ-related information and conducted several studies that are useful in the reconsideration of EPZ size and planning elements for SMRs and the associated process for regulatory change:

- SECY-97-020 (Reference 6.1.8), which provides results of a staff evaluation of emergency planning for large advanced LWRs.
- Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs, SECY-10-0034; SECY-11-0152, and SECY-15-0077 (Reference 6.2.10, Reference 6.1.6, and Reference 6.2.3, respectively), which discuss the staff's intent to develop a framework for SMR emergency planning, address associated policy and technical issues, and present options for revising emergency planning regulations and guidance for SMRs that are discussed further in Section 2.3.
- Criteria for Protective Action Recommendations for Severe Accidents, Volumes 1, 2, and 3, NUREG/CR-6953 (Reference 6.2.11), which evaluates the efficacy of various protective action strategies within the EPZ.

- Identification and Analysis of Factors Affecting Emergency Evacuations, Volumes 1 and 2, NUREG/CR-6864 (Reference 6.2.12), which examines actual evacuations carried out in the U.S., in response to natural disasters and man-made, nonnuclear incidents, between 1990 and 2003 to gain a fuller understanding of the dynamics involved.
- Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations, NUREG/CR-6981 (Reference 6.2.13), which assesses Hurricanes Katrina, Rita, and Wilma, as well as other large scale evacuations, for lessons learned to further enhance the emergency preparedness program for radiological emergencies at nuclear power plants.
- State-of-the-Art Reactor Consequence Analysis (SOARCA) Report, NUREG-1935, (Reference 6.2.14), which evaluates fission product releases, associated off-site consequences, and hypothetical evacuations in response to potential accidents in operating plants.

2.2 NEI White Paper

The white paper (Reference 6.1.5) describes a generic methodology and criteria for establishing the technical basis associated with plume exposure SMR EPZ sizing. The white paper is in support of the continuing dialogue with the NRC on emergency preparedness and SMR-appropriate plume exposure EPZ size, and responds to SECY-11-0152 (Reference 6.1.6), which discusses the NRC staff's intent to develop an emergency planning framework for SMRs. The paper addresses SMRs with light-water-cooled and moderated designs only, and is not applicable to other types of SMRs. The white paper indicates that the technical basis for determining the EPZ size that is appropriate for SMRs is rooted in their enhanced safety. This technical basis recognizes and allows for what is expected to be reduced risk and increased safety margins of the SMR designs, including smaller cores, decreased likelihood of accidents, and smaller, slower, fission product releases in the unlikely event of an accident.

At a high level, the paper is a first step in developing a methodology for establishing the technical basis for determining EPZ size. It proposes a risk-informed approach with two complementary efforts: (1) using the plant-specific probabilistic risk assessment (PRA) to inform EPZ sizing considerations; and (2) providing enhanced plant capabilities to account for uncertainties, including an operationally-focused mitigation capability in support of the defense-in-depth philosophy.

2.3 NRC EPZ-Related Rulemaking Documents

In SECY-11-0152 (Reference 6.1.6), the staff informed the Commission of their intent to develop a technology-neutral, dose-based, consequence-oriented emergency planning framework for SMR sites. The staff found that “it may be appropriate for SMRs to develop similarly reduced EPZ sizes, commensurate with their accident source terms, fission product releases, and accident dose characteristics.” The projected approach for EPZ sizing was based on accident dose limits at the EPZ boundary, and would include “establishing criteria for determining the point at which the probability of exceeding the PAG is acceptably low.”

In SECY-15-0077 (Reference 6.2.3), the staff sought Commission approval to revise NRC regulations and guidance through rulemaking to require SMR license applicants to demonstrate how their proposed facilities achieve appropriate dose limits at the specified EPZ distance, which may be as low as the site boundary. The SECY indicates that the regulations can be established generically without site- or design-specific information regarding source term, fission products, or projected off-site dose. The staff anticipates that the technical basis for the regulations would be developed as part of rulemaking. This would include quantitative guidelines and criteria for accident selection and evaluation, and would be applicable to SMRs but not to operating plants. The NRC will review design and licensing information provided by SMR applicants to ensure that the off-site dose consequences are commensurate with the requested EPZ size and to ensure that applicable requirements for adequate protection of public health and safety, and the environment, are met. In the staff requirements memorandum (SRM) associated with SECY-15-0077, dated August 4, 2015 (Reference 6.1.7), the Commission approved the staff's recommendation to initiate the rulemaking.

In Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies, SECY-16-0069 (Reference 6.2.15), the staff outlines the proposed rulemaking timeline for the change to the emergency planning rules for SMRs in 10 CFR 50.47 and 10 CFR 50, Appendix E. SECY-16-0069 proposes an estimated schedule beginning in August 2016 with the initiation of the regulatory basis phase. In the SRM associated with SECY-16-0069, dated June 22, 2016 (Reference 6.2.16), the Commission approved the staff's proposed schedule.

The NRC staff released a regulatory basis to initiate a rulemaking to revise regulations and guidance for emergency preparedness for SMRs and other new technologies, Rulemaking for Emergency Preparedness for Small Modular Reactors and Other New Technologies: Regulatory Basis (Reference 6.1.10). The regulatory basis document examines the existing emergency planning regulatory framework, anticipated regulatory issues, potential regulatory approaches, other regulatory considerations (such as cost and impact considerations), stakeholder interactions, and the next steps towards rulemaking and guidance documents. The NRC staff recommends providing rules and guidance focusing on establishing EPZ requirements for SMRs and other new technologies based on the principles and methodology outlined in NUREG-0396.

In December 2019, the NRC granted an ESP to the Tennessee Valley Authority (TVA) for the Clinch River Site (CLI-19-10, Reference 6.2.17). Chapter 13 of the TVA site safety analysis report describes a methodology for determining a plume exposure EPZ distance for an SMR installation at the Clinch River Site (Reference 6.2.18). The NRC staff findings summarized in Section C.2 of CLI-19-10 show that the basis for a reduced EPZ size in the TVA ESP is acceptable because it maintains the same level of protection as large LWR EPZs. Additionally, the Commission "[does] not view TVA's proposal [of site boundary EPZ] as eliminating an element of defense in depth; rather, emergency planning activities would be appropriately scaled to reflect the potential hazards posed by the facility."

In May 2020, the proposed rule for Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or

Utilization Facilities and accompanying draft guidance were released for public comment (Reference 6.2.19 and Reference 6.2.20). The proposed rule would replace the option for case-by-case basis EPZ size determinations for reactors with an authorized power level less than 250MWt with the option to determine the plume exposure pathway EPZ for an SMR or other new technology as the area within which the dose to an individual is projected to exceed 1 rem TEDE over an exposure time of 96 hours from the release of radioactive materials resulting from a spectrum of accidents for the facility.

2.4 NuScale Approach

The NuScale approach for technical justification of EPZ size is based on the NEI white paper, NRC feedback on the white paper, and key NRC EPZ-related documents such as NUREG-0396 and associated NRC staff licensing reviews (Reference 6.1.11 and Reference 6.1.12), SECY-97-020, SECY-11-0152, and SECY-15-0077. It incorporates experience and lessons learned from risk-informed decision-making in regulatory applications. The NuScale approach uses a risk-informed evaluation of severe accidents, which balances risk considerations and defense-in-depth.

Key elements of the NuScale approach include:

- Identification of appropriate accident sequences based on accident sequence frequency information from the PRA in order "to determine appropriate accidents to be evaluated" (Reference 6.2.3) for the EPZ basis.
- The PRA used in the application of this methodology must be of appropriate scope and technically acceptable. Regulatory Guide (RG) 1.200 describes one approach for determining the acceptability of a PRA (Reference 6.2.21). The RG provides guidance on the four areas that collectively determine the acceptability of a PRA (i.e., scope, technical elements, level of detail, and plant representation) that can be met using national consensus PRA standards and a peer review.
- Determination of whether accident sequences are less severe or more severe, distinguished by containment integrity.
- Use of dose-based criteria and a consequence orientation, consistent with NRC guidance and applicable historical concepts for EPZ development, as discussed throughout this report.
- Use of state-of-the-art tools, supported by four decades of severe accident research and methods development, in the analytical evaluation of source terms and dose consequences for accident sequences, which are screened in to the EPZ technical basis.
- Consideration of all internal and external hazards and all operating modes, including low power and shutdown, multi-module accidents, and other risks such as spent fuel pool (SFP) accidents (when applicable) to provide assurance of completeness.
- A qualitative, plant-level evaluation of defense-in-depth to account for PRA uncertainties.
- A methodology to assess severe accident modeling uncertainties as confirmation of analytical results.

2.5 Conditions of Applicability

The conditions of applicability must be satisfied by an applicant in order to implement this EPZ methodology.

2.5.1 LWR Condition

The following condition applies to the LWR design when implementing the EPZ methodology:

The LWR is a NuScale SMR design, including the standard plant design (Docket 52-048) and variations and derivatives thereof comprising all of the following characteristics:

a. small modular integral pressurized LWRs, meaning reactor modules composed of a reactor core, primary cooling loop, pressurizer, and steam generator(s) within a reactor vessel, housed within a containment vessel normally operated at subatmospheric pressure conditions,

b. operating modules partially immersed in water that serves as the UHS,

c. the UHS is retained below grade in a structure with up to 12 reactor modules per UHS,

d. a safe shutdown earthquake with a peak ground acceleration of 0.5g, and

e. structures, systems, and components (SSCs) capable of performing their safety functions without AC electric power, DC electric power, or operator actions for at least 72 hours following a design basis event.

2.5.2 PRA Conditions

The following conditions apply to the PRA used in implementing the EPZ methodology:[1]

Condition 1: The PRA addresses internal and external hazards and all operating modes.

Condition 2: The PRA is demonstrated to be technically acceptable for this purpose.[2]

1. Applicants who wish to utilize the EPZ methodology should identify this use of the PRA in Chapter 19 of the final safety analysis report.
2. It is recognized that some requirements are not feasible for a new nuclear power plant design (e.g., a new design, by definition, will not have operational experience until it has been licensed, constructed, and operated; a plant walk down is not possible if the plant has not yet been constructed). As described in RG 1.200, not all requirements need to meet current good practices (i.e., Capability Category II of an ASME/ANS PRA standard) for all applications.

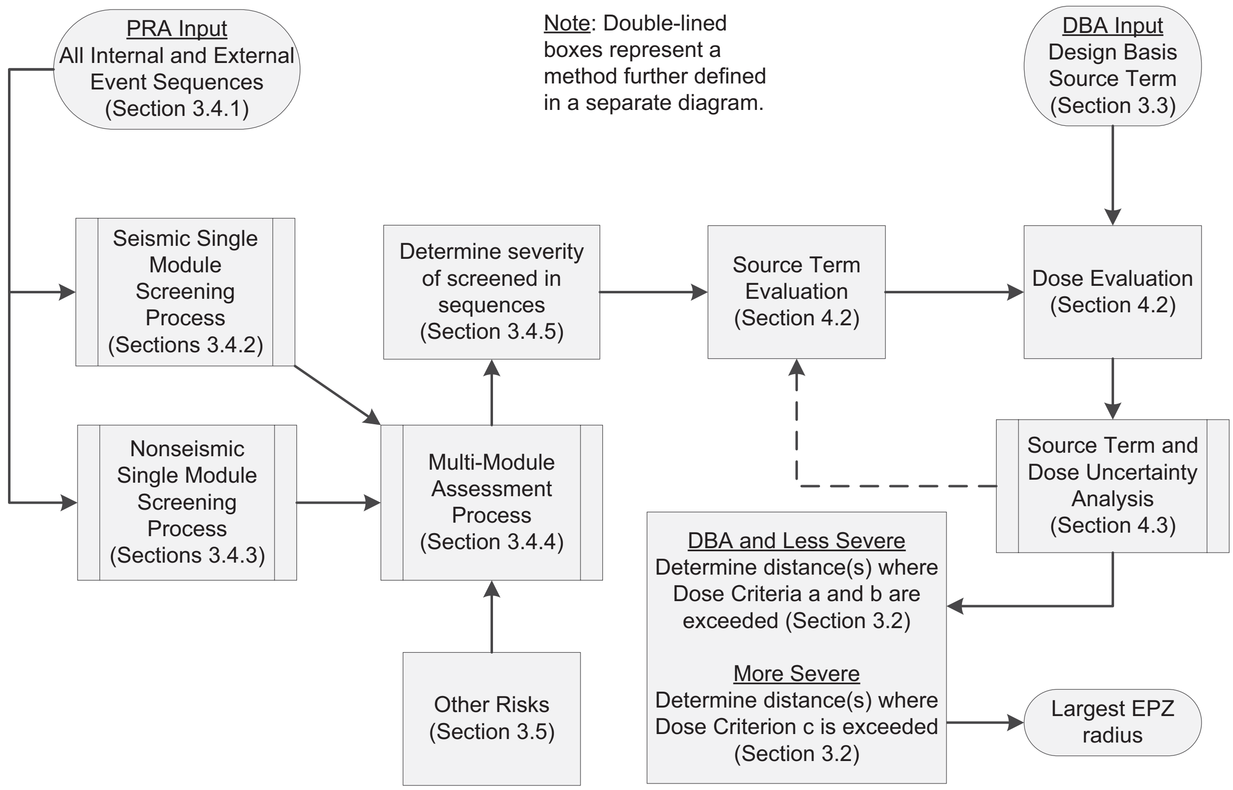
3.0 Accident Screening Methodology

Section 3.0 of the LTR addresses the methodology for determining appropriate accident sequences to be evaluated for the plume exposure EPZ basis. Section 3.1 presents key assumptions and Section 3.2 discusses the dose-based criteria for EPZ sizing. Sections 3.3 and 3.4 discuss the methodology for using a risk-informed approach to select appropriate single and multi-module accident sequences to include in the EPZ technical basis. Sections 3.5 and 3.6 discuss the methodology for addressing other PRA risks and security events, respectively. Section 3.7 includes a qualitative, plant-level evaluation of defense-in-depth which conforms to regulatory guidance and past NRC practice. Section 3.8 includes a review of key assumptions and sources of uncertainty in the underlying PRA. Figure 3-1 provides an overview of the EPZ methodology, as well as identification of the section where more information can be found. Each step that will be implemented by the COL applicant to determine the final EPZ distance is also presented below with corresponding LTR section number:

- Compile accident sequences from the PRA for all internal and external initiators (Section 3.4.1)
- Perform screening of seismic accident sequences based on {{ }}2(a),(c) (Section 3.4.2)
- Perform screening of nonseismic single module accident sequences based on frequency, including uncertainty (Section 3.4.3)
- Perform multi-module screening for multi-module sites, including uncertainty (Section 3.4.4)
- Characterize screened-in sequences as more or less severe (i.e., containment bypassed or intact) (Section 3.4.5)
- Identify and evaluate other risks with an impact to EPZ sizing (Section 3.5)
- Perform severe accident simulations of screened-in accident sequences to determine environmental source term (Section 4.1.1 and 4.2)
- Add the design-basis source term (DBST) (Section 3.3), which is a surrogate less severe accident
- Perform consequence simulations with MELCOR Accident Consequence Code System (MACCS), using the severe accident source terms and the DBST (Sections 4.1.2 and 4.2)
- Confirm MACCS results within 0.5 km of the radiological release location, as necessary (Section 4.2.4)
- Perform uncertainty analysis on severe accident and consequence modeling and justify important parameters (Section 4.3)
  - Repeat severe accident and consequence simulations if necessary

- Perform plant-level, qualitative evaluation of defense-in-depth to confirm that the design features and safety strategy employs successive compensatory measures to prevent accidents or mitigate consequences (Section 3.7)
- Perform a review of the underlying PRA to address key assumptions and uncertainty and their impact on the application of the EPZ methodology (Section 3.8)
- Determine the final EPZ distance as the largest distance among the following:
  - The distance at which dose does not exceed either a 1 rem TEDE criterion at mean weather conditions or a 5 rem TEDE criterion at 95th percentile weather conditions for DBST (Section 4.2.1)
  - The distance at which dose does not exceed either a 1 rem TEDE criterion at mean weather conditions or a 5 rem TEDE criterion at 95th percentile weather conditions for screened-in less severe accident sequences (Section 4.2.2)
  - The distance at which {{ }}$^{2(a),(c)}$ screened-in more severe accident sequences (Section 4.2.3)
  - The site boundary (i.e., the minimum EPZ)

Figure 3-1 Overall methodology to determine EPZ distance



3.1 Assumptions

The following assumptions are justified to form a basis for the EPZ sizing methodology:

Assumption 1: Risk-informed methods are appropriate for EPZ sizing.

Justification: Risk-informed methods and applications have progressed to the point where they provide an appropriate framework to determine EPZ sizing. Important aspects of this progress in risk-informed methods and applications include:

- PRA development and advancement, including plant-specific PRAs performed by licensees and NRC over 20 plus years, the SOARCA study (Reference 6.2.14), and evolution of PRA industry-consensus standards (e.g., Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications [Reference 6.3.1]) including standards that have been or are being reviewed with the expectation of ultimately being endorsed by the NRC.

- Evolution of risk-informed applications, including risk-informed changes to the licensing basis for operating plants (i.e., An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, RG 1.174 [Reference 6.3.2]), the reactor oversight process, risk-informed categorization and treatment of structures, systems, and components (i.e., 10 CFR 50.69, Reference 6.3.12), and new reactor licensing where all new designs are required to perform a PRA and apply PRA insights to the design process.
- The Risk Management Task Force formed at Commission direction in 2011, culminated with issuance of A Proposed Risk Management Regulatory Framework, NUREG-2150 (Reference 6.3.3), which described a proposed framework for risk management that would provide risk-informed and performance-based defense-in-depth protections.
- The Fukushima Near-Term Task Force issued Recommendations for Enhancing Reactor Safety in the 21st Century (Reference 6.3.4), which recommended establishing a regulatory framework for adequate protection that balances defense-in-depth and risk considerations.
- A risk-informed approach to Emergency Action Levels (i.e., Risk Informing Emergency Preparedness Oversight: Evaluation of Emergency Action Levels-A Pilot Study of Peach Bottom, Surry, and Sequoyah, NUREG/CR-7154, Reference 6.3.5).

Recent NRC SECYs addressing SMRs also reflect this progress in risk-informed methods and applications:

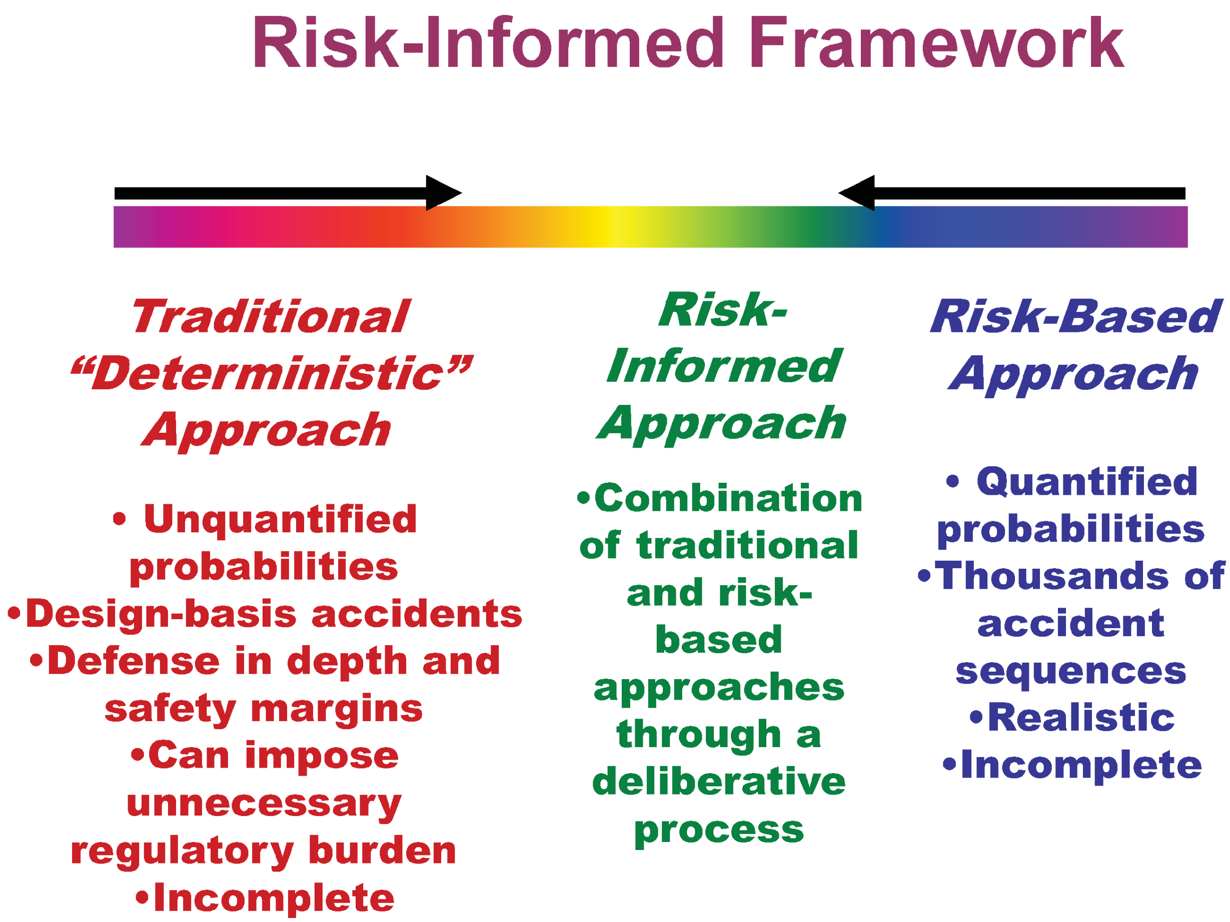
- SECY-10-0034 (Reference 6.2.10) states that the NRC staff plans to use a risk-informed and performance-based approach that employs deterministic judgment and analysis complemented by PRA information to review design and license applications for SMRs.
- SECY-11-0152 (Reference 6.1.6) states that an appropriate method for addressing EPZ size would involve using a PRA that includes dose assessment, which is based on current insights in severe accident progression.
- SECY-15-0077 (Reference 6.2.3) states that the concept of EPZ size commensurate with the off-site radiological risk is not new to the NRC.
- The recent NRC regulatory basis for SMR emergency preparedness rulemaking (Reference 6.1.10) states that the staff is going to be using a risk-informed approach for selection of licensing basis events and also that the proposed rulemaking will enhance the risk-informed regulatory framework.

<u>Assumption 2</u>: In the risk-informed approach to EPZ sizing, consideration is given to both defense-in-depth and quantitative risk results.

<u>Justification:</u> Risk-informed processes for any regulatory application, in particular EPZ sizing, should combine and balance insights from deterministic and probabilistic insights. This is illustrated in Figure 3-2, which shows a risk-informed approach as a combination of a traditional, "deterministic" approach, and a risk-based approach, as presented in NRC Risk-Informed and Performance Based Initiatives slides (Reference 6.3.6). A qualitative evaluation of defense-in-depth, consistent with regulatory guidance and

practice, is included in the NuScale risk-informed approach to confirm the existence, functionality, and capability of design features and strategies that balance accident prevention and mitigation in order to provide confidence in the acceptably low plant risk and demonstrate protection of the health and safety of the public.

Figure 3-2 Risk-informed framework



Addressing EPZ using a risk-informed approach offers an opportunity to optimize EPZ size and its basis using a more balanced, transparent process. In the generic process used in NUREG-0396 in the 1970s, the margins of safety provided by the EPZ were based on a combination of risk insights from WASH-1400 and "were qualitatively found adequate as a matter of judgment" (Reference 6.1.8). Given the high level of safety achieved with light water SMR designs, as demonstrated by significant margin to the NRC safety goals, this qualitative, generic concept for determining the adequacy of the margins of safety needs to be updated to include a risk-informed, design-specific approach where there is appropriate consideration given to both quantitative and qualitative methods. In the time since NUREG-0396 was published, the severe accident experimental knowledge base and analytical methods have advanced to the point that tools and models are now available to support this methodology for EPZ sizing.

<u>Assumption 3</u>: A dose-based approach with a consequence orientation is appropriate for the EPZ sizing basis.

<u>Justification</u>: The risk-informed approach is implemented in a way that addresses NRC guidance and applicable, historical concepts for EPZ development. This includes:

- applying a dose-based framework with a consequence orientation (SECY-15-0077)
- events should provide an acceptable spectrum of consequences (SECY-11-0152)
- use of a "spectrum of accidents as a basis for developing emergency response plans" (NUREG-0396)
- use of a "spectrum of credible accidents for the facility" as the basis for EPZ size (SRM-SECY-18-0103)

As noted in the NEI white paper (Reference 6.1.5), industry experience indicates that attempts at applying quantitative, PRA-based information in decision-making on regulatory matters have been challenging. Perceived uncertainties associated with state of knowledge limitations and with hazards and events not easily amenable to treatment in a PRA have often led to overly conservative solutions and unrealistic assumptions about postulated accidents. The risk-informed approach includes steps to achieve a consequence orientation without resorting to such solutions. These steps include:

- design and operational features that provide multiple, independent layers of defense-in-depth and very low accident sequence frequencies with consideration of uncertainty
- use of state-of-the-art methods to calculate source terms and doses, which greatly reduce uncertainty as compared to previous quantitative methods, that were excessively conservative based on the state of knowledge and methods at that time
  - integrated uncertainty analysis is also used to increase confidence in the best-estimate source term and consequence results as discussed in Section 4.3
- in addition to the state-of-the-art quantitative methods, application of qualitative means to address uncertainties in the context of very low frequency events including:
  - requirement to assess off-site dose consequences from the conservative DBST (Section 3.3)
  - a plant-level, qualitative evaluation of defense-in-depth to demonstrate adequate balance between accident prevention and mitigation of potential consequences as an extension to emergency planning and NRC's existing defense-in-depth philosophy and guidance
  - required site emergency plans will provide a base for expanding response efforts, if necessary, in accordance with regulatory guidance so as to provide an additional layer of defense-in-depth

The aforementioned steps preclude the need for selection of unnecessarily conservative solutions as part of achieving a consequence-orientation in the EPZ sizing basis.

3.2 Dose-Based Criteria

Applicants shall demonstrate acceptability by use of dose-based criteria, based on NUREG-0396, which industry considers appropriate for a risk-informed approach to EPZ plume exposure sizing. The dose criteria employed in this methodology are:

- Criterion a: The EPZ should encompass those areas in which projected dose from DBAs could exceed the early phase PAGs.
- Criterion b: The EPZ should encompass those areas in which consequences of less severe accident sequences could exceed the early phase PAGs.
- Criterion c: The EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe accident sequences.

The early phase PAGs for Criteria a and b are 1 and 5 rem[3] TEDE (Reference 6.1.4). Throughout the remainder of the report, whenever the term "PAGs" is used, it refers only to the early phase EPA PAGs. While the application of Criterion a is consistent with NUREG-0396, application of Criterion b is conservative. In NUREG-0396, beyond-design-basis accidents are initially compared to a 200 rem whole body acute dose threshold (i.e., Figure I-11); then as a confirmatory step, less severe beyond-design-basis accidents are compared to the PAGs (i.e., Figure I-15). In contrast, this methodology conservatively uses the PAGs exclusively for less severe accidents.

The metric used in Criterion c for substantial reduction in early severe health effects is 200 rem whole body acute dose as indicated in NUREG-0396. In this application, red marrow acute dose is used as an acceptable surrogate for whole body acute dose, as discussed in Section 4.2. The methodology for determination of appropriate accident sequences to be evaluated against the criteria is addressed in Sections 3.3 and 3.4, and method details for applying the dose criteria are provided in Section 4.2. To satisfy Criterion c, the {{

}}2(a),(c) at the EPZ boundary.

{{

}}2(a),(c)

3.The EPA PAGs establish dose limits as a range from 1 to 5 rem. These are discussed in the context of the doses in which evacuation (when projected doses exceed 1 rem) and sheltering in place (when projected doses exceed 5 rem) represents less risk to the public than radiological exposure. This LTR conservatively establishes two dose acceptance criteria: 1 rem TEDE for mean meteorology and 5 rem TEDE for 95 percent meteorology. 1 rem is the primary criterion, as it represents a lower dose limit and conforms with the EPA recommendations for best-estimate modeling. 5 rem is also a criterion that will be met but it is mainly a confirmation of the results of the primary criteria for unlikely weather conditions that could cause increased dose over a limited area.

{{

}}2(a),(c)

3.3 Determination of Appropriate Design-Basis Accidents to Be Evaluated

For Criterion a (Section 3.2), the methodology requires evaluation of the DBST from Chapter 15 of the COL applicant's final safety analysis report (FSAR). The DBST is a surrogate release from containment used to assess off-site dose. The release from the containment is conservatively assumed to go directly into the environment with no credit for holdup or deposition between the containment and the environment. The DBST will be evaluated for the Criterion a comparison against the PAGs as discussed in Section 4.2. The EPZ distance as calculated by the DBST will be compared to the EPZ distances from the more and less severe accident sequences screened in Section 3.4. The final EPZ distance is the largest of these distances.

It is noted that "source term" in the context of the EPZ sizing source term evaluation methodology refers to fission product release to the environment as a function of time. Historically, "source term" has been used to refer to fission product release into containment, such as the source term in RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Reference 6.3.7). The DBST in Chapter 15 includes a release into containment, and will most likely evaluate fission product transport and deposition within containment to determine the release to the environment. However, if the fission product behavior within the containment is conservatively neglected in the Chapter 15 analysis, the COL applicant may have to extend the DBST analysis to support the EPZ methodology.

3.4 Determination of Appropriate Beyond-Design-Basis Accidents to Be Evaluated

For Dose Criteria b and c (Section 3.2), the methodology of this section is used to screen beyond-design-basis accident sequences for inclusion in the EPZ technical basis. PRA accident sequences from all internal events, external events, and operating modes are first identified. Screening criteria are then applied to identify a spectrum of accident sequences for EPZ consideration (Sections 3.4.2, 3.4.3, and 3.4.4). Once the spectrum of sequences is determined, the evaluation of these sequences involves calculating a source term and dose, as discussed in Sections 4.2.2 and 4.2.3, using the dose criteria in Section 3.2.

There are three main elements of the method to identify the spectrum of accidents: (1) initial sequence compilation (described in Section 3.4.1), (2) accident sequence screening (described in Sections 3.4.2, 3.4.3, and 3.4.4), and (3) final classification of severity (described in Section 3.4.5).

A multi-module process for designs with multiple co-located modules is detailed in Section 3.4.4.

3.4.1 Compilation of Probabilistic Risk Assessment Accident Sequences

To support the identification of applicable accident sequences, PRA accident sequences for all internal events and external events, overall operating modes, are compiled. The use of a PRA as described in Section 2.5 and all associated accidents is appropriate to identify applicable accident sequences to be considered in the EPZ methodology.

A point estimate sequence-level core damage frequency (CDF), which is calculated to approximate the mean of a distribution, is appropriate for a best-estimate EPZ evaluation. The EPZ methodology incorporates a separate uncertainty analysis, which supports the use of a best-estimate evaluation.

In the PRA, a "sequence" refers to the progression from initiating event to an end state within an event tree, with each sequence representing a unique accident progression. All sequences are treated individually within the accident selection process of the EPZ methodology. Sequences used in the screening process are expected to be defined by an initiating event and top events representing the success or failure of mitigating systems at the system level. Any combination of failures of individual components or subcomponents within a system that fail the system mission in a top event are considered to belong to the same sequence. An accident sequence is a common element of a PRA; it is defined and governed by specific high level and supporting requirements in the ASME/ANS Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (Reference 6.3.1).

Use of individual sequences also removes ambiguity from performing source term and dose analyses. However, screened-in sequences may be grouped into release categories to reduce the number of required source term and dose consequence simulations as discussed in Section 4.0. The applicant must identify and justify the technical basis for the grouping of sequences into release categories, if used.

Anticipated transient without scram (ATWS) events are treated in the EPZ methodology in the same manner as reactor trip (non-ATWS) events. That is, the PRA event trees have a top event where success is reactor trip and failure is ATWS. Therefore, there are unique sequences for ATWS and non-ATWS that are considered separately.

The frequency of beyond-design-basis event PRA sequences often includes operator actions that are consistent with both generic technical guidelines and emergency operating procedures (EOPs). Thus, these types of operator actions are included in the PRA event trees and reflected in accident sequence frequencies. Any mitigation beyond the EOPs (i.e., severe accident management guidelines [SAMGs] and extensive damage mitigating guidelines [EDMGs]), however, should not be credited in frequency screening, as the probabilities of these human actions have historically been difficult to quantify. This results in a conservative sequence frequency.

3.4.2 Screening of Seismic Single Module Accident Sequences {{ }}2(a),(c)

The relatively large uncertainty compared to internal hazards captured in the methods used to predict the seismic hazard likely results in a large amount of conservatism in the prediction of both the seismic occurrence rate and the severity of the local acceleration forces. In the methods used much of this uncertainty is characterized as being random (i.e., aleatory) and hence irreducible. The probability distributions used in seismic analysis methods are lognormally shaped (i.e., skewed to higher, more conservative values). As such, the relatively large uncertainty compared to internal hazards characterized by these uncertainty distributions will pull the mean values and upper bounds into the more conservative end of the distribution. Because of these uniquely large uncertainties, seismic risk is assessed separately from nonseismic risks in the EPZ methodology.

This section describes the treatment of seismic accident sequences within the overall screening process shown in Figure 3-1. Applicants demonstrate acceptable treatment of the seismic hazard by {{

}}2(a),(c)

Seismic sequences contain the characteristics of PRA sequences described in Section 3.4.1, with one clarification: {{

}}2(a),(c)

The screening criteria outlined above provide a balanced approach to the consideration of seismic sequences initiated by relatively frequent low intensity earthquakes (with consideration of their potential consequences), as well as consideration of larger and rarer earthquakes that impose more extreme demands on nuclear power plant structures and infrastructure, with more catastrophic consequences.

{{

}}2(a),(c)

Figure 3-3 {{ }}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

{{

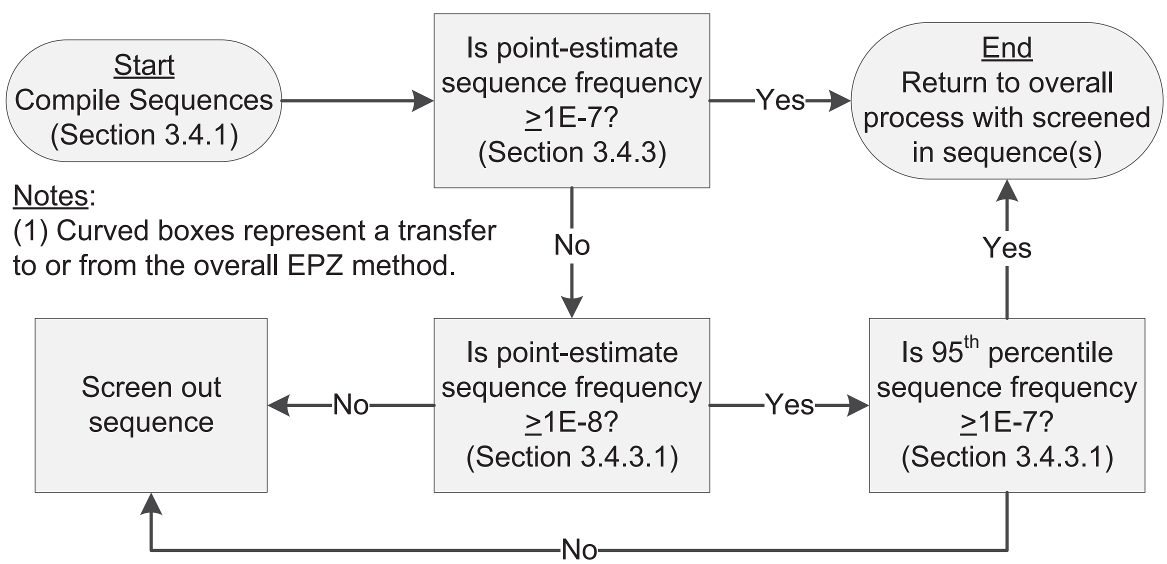
}}2(a),(c)

3.4.3 Screening of Nonseismic Single Module Accident Sequences on Core Damage Frequency

This section describes the screening process for all nonseismic accident sequences. This screening is based on the sequence CDF; all nonseismic core damage sequences with a frequency greater than or equal to 1E-7 per module year are screened into the EPZ sizing analysis. Parametric uncertainty in initially screened-out sequences is evaluated to ensure the complete set of source terms is retained in the spectrum of accident sequences. Figure 3-4 provides an overview of the nonseismic sequence screening process. It is noted here that the sequence CDFs that are compared to the screening criteria are to be point estimates that are calculated using mean values for the PRA model basic events. The use of mean values for PRA basic events is typically done in PRA quantification, but the calculated sequence CDF is termed a point estimate since this calculation result will typically differ by a small amount from the mean value CDF that results from a full uncertainty analysis of the PRA. Applicants demonstrate acceptability by:

- screening in all nonseismic core damage accident sequences with a point estimate sequence frequency ≥ 1E-7/module year
- evaluating parametric uncertainty in nonseismic core damage accident sequences with a point estimate sequence frequency near the 1E-7/module year threshold

Figure 3-4 Nonseismic Sequence Screening Process



All frequencies in this section are per module year to ensure consistency with past precedent of per reactor year in NUREG-0396 (Reference 6.1.3). It is also noted that the use of per module year in the screening process does not impact the treatment of multi-module accidents. Multi-module accidents are addressed in Section 3.4.4.

All nonseismic core damage sequences with a frequency greater than or equal to 1E-7 per module year will be retained for analysis in the EPZ technical basis. This screening criterion is consistent with the spectrum of accidents that form the basis for the original EPZ sizing in NUREG-0396, capturing the range of WASH-1400 release category frequencies. Additionally, identifying the spectrum of accident sequences based on CDF conservatively ignores the conditional probability of radionuclide release when compared to the quantification of WASH-1400 release category frequencies.

The sequence CDF screening threshold of 1E-7 per module year likely represents at least an order-of-magnitude safety margin compared to the NRC safety goal of a large release frequency being less than 1E-6 per year (assuming a conditional large release probability of less than 10 percent). As documented in SECY-93-138, the NRC staff has concluded that limiting the large release frequency to 1E-6 per year (or less) results in a degree of conservatism several orders of magnitude more conservative than the early health effects safety goal.

3.4.3.1 Parameter Uncertainty in Nonseismic Sequence Screening

Applicants demonstrate acceptability by evaluating parameter uncertainty in the context of screening based on sequence frequency to capture a complete spectrum of source terms in the EPZ basis.

Parameter uncertainty in the nonseismic sequence screening is addressed consistent with NUREG-1855 (Reference 6.3.9) as follows:

1. The nonseismic core damage sequence frequencies will be calculated as point-estimates that are approximations[4] of mean values.
2. Identify the proximity of the point-estimate sequence frequency to the 1E-7 per module year screening criteria.
3. If the point-estimate sequence frequency is close to the screening threshold, then the sequence upper bound is compared to the screening criteria.
4. Consider for inclusion to the EPZ sizing method those sequences that challenge the screening criteria in the EPZ basis.

In the specific context of the EPZ sizing method, these steps take the following form:

1. If the point-estimate value of the sequence is an order of magnitude or more below the screening criteria (i.e., less than or equal to 1E-8 per module year), the sequence is screened-out without further consideration.
2. If the point-estimate value of the sequence is within an order of magnitude of the screening criteria, then the 95th percentile value will be compared to the

4. Note that in this context a point-estimate is calculated using the mean values for the PRA basic events in a point-estimate calculation (in contrast to an uncertainty analysis that will generate a full probability distribution on the quantification of the sequence frequencies). The point-estimate calculated in this manner will typically differ by a small amount from the mean value of the probability distribution generated by the uncertainty analysis.

screening criterion. If the 95th percentile value is also below the screening criterion, then the sequence is screened-out of the process.

3. If the point-estimate value of the sequence is within an order of magnitude of the screening criteria and the 95th percentile value is greater than or equal to the screening criteria, then the sequence is screened-into the process and included in the source term and dose analysis.

3.4.4 Multi-Module Accident Methodology

This section applies to designs that feature multiple co-located reactor modules. Applicants demonstrate acceptability by retaining multi-module accident sequences that meet the following criteria, as shown in Figure 3-5:

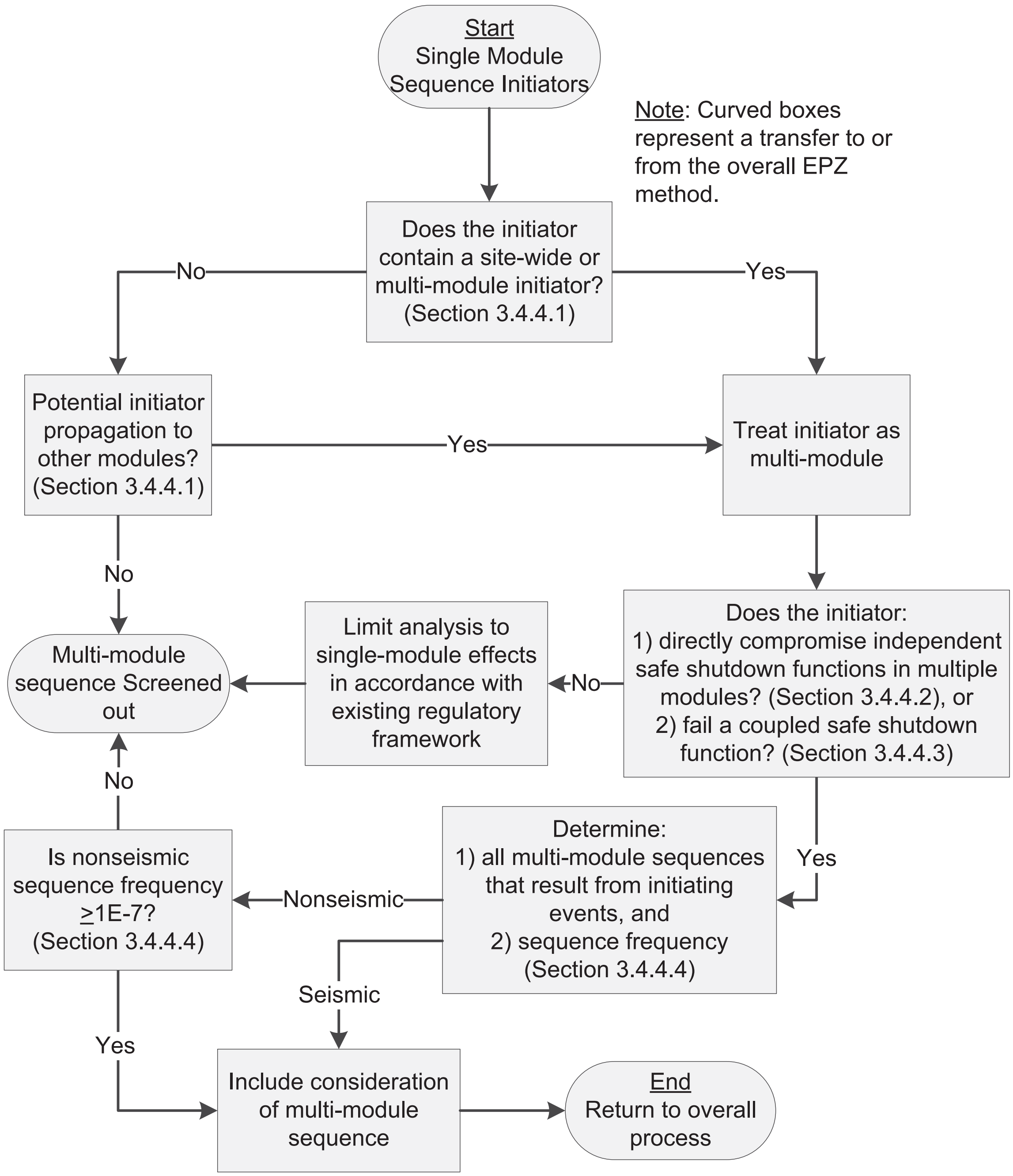
- The sequence contains a site-wide or multi-module initiator, or a single-module initiator that physically propagates to other modules.
- Safe shutdown mechanisms are directly compromised in multiple modules, or a coupled safe shutdown function is failed.
- The resultant nonseismic multi-module accident sequences have a CDF greater than or equal to 1E-7/module year, including evaluation of parametric uncertainty.

Mechanisms that can cause damage to a single module are analyzed for their potential to propagate to other modules. Only sequences that are screened in by the methodology in Section 3.4.3 for a single module are subjected to additional multi-module screening. Regardless of CDF, seismic sequences screened in by the criteria outlined in Section 3.4.2 are not subject to additional multi-module sequence screening. All multi-module seismic sequences identified in this section are retained for further source term and dose consequence analysis.

Several aspects of accident sequences have potential multi-module effects. The following aspects are considered in the assessment process shown in Figure 3-5:

- Initiating events - Initiators may affect multiple modules by definition, such as loss of off-site power or a failure of a shared support system. Alternatively, coupling mechanisms may exist that increase the likelihood of an initiator occurring in more than one module. An example of the latter would include a spurious reactor trip caused by faulty instrumentation of more than one module.
- Correlated failures in mitigating functions following multi-module initiators - Examples include valves failing to open when required in multiple modules due to a common failure mechanism, functions in separate modules being similarly affected by a common initiator (e.g., an external event), or failures in shared mitigating systems. However, the failures of independent mitigating systems that are simultaneously tripped by the initiating event are not considered correlated.
- Human actions - Actions performed on separate modules can have a degree of dependency. Several systems may be shared in whole or in part among multiple modules, consequently the results of the associated human actions are dependent on shared equipment availability and available operator resources.

Figure 3-5 Multi-module assessment process



Section 3.4.4.1 describes the multi-module implications associated with initiating events. Section 3.4.4.2 assesses correlated failures between modules. Section 3.4.4.3 assesses shared system failures. These sections detail the specific considerations that will be evaluated to follow the process shown in Figure 3-5.

3.4.4.1 Hazards and Initiating Events

The evaluation of multi-module accidents requires that all initiators from screened-in accident sequences be assessed against the criteria described in Figure 3-5. These hazards have the potential to include:

- internal events
- internal fires
- internal floods
- high winds
- external floods
- seismic events

Depending on which sequences screen in, the assessment may include all operating modes including full power, low power, and shutdown.

Potential multi-module implications and coupling mechanisms are identified. Multi-module implications and coupling mechanisms are those susceptibilities (through shared systems or common areas) to a particular hazard. Specifically, the potential for random failures to occur simultaneously across multiple modules is not a design-specific concern. Accident sequences that require random failures to occur in multiple modules in addition to any hazard-induced component failures are screened from further consideration. For seismic events, while the initiator will impose demands on all equipment and modules simultaneously, the consequence of each failure is reviewed for multi-module implications, including seismically-induced falling and interaction hazards that may impact multiple modules. This process is explored further as part of correlated failures addressed in Section 3.4.4.2.

3.4.4.2 Correlated Failures between Modules

There is a potential for the initiating event to induce failures that are correlated between modules. For example, there can be a common failure mechanism of identical but independent systems that are demanded for multiple modules or a structural failure caused by the initiating event could impact multiple modules. In accordance with the process described in Figure 3-5, if the initiator can induce any correlated failures of independent systems, any multi-module impacts will be addressed.

3.4.4.3 Shared System Failures

Plant systems may be shared in whole or in part among multiple modules, meaning their failure could have multi-module impacts. In accordance with the process described in Figure 3-5, if the initiator involves a shared system, or impacts a shared mitigating function between modules, any multi-module impacts will be addressed.

3.4.4.4 Multi-Module Conclusions

3.4.4.4.1 Determination of Multi-Module Accidents

The application of this methodology results in the identification of initiating events from screened-in single module sequences that can lead to a multi-module accident sequence. There are multiple potential accident sequences for each initiating event; therefore, all sequences that result in a multi-module accident will be determined. Following identification of each sequence, the CDF of each multi-module sequence is determined, which incorporates the likelihood of core damage in each impacted module.

As a final step, a 1E-7 per module year CDF screening criterion (Section 3.4.3) and consideration of parameter uncertainty (Section 3.4.3.1) is again used to screen nonseismic multi-module sequences based on their calculated CDFs. This method, as shown in Figure 3-5, is used to consider nonseismic multi-module impacts for EPZ.

All seismic sequences that are included in the EPZ technical basis are evaluated for multi-module impacts. {{

}}2(a),(c)

Multi-module sequences that are included in the EPZ technical basis require source term and dose analyses. The number of affected modules for each screened-in sequence will be determined by the mechanism that leads to core damage and/or large release. It is noted that the potential release paths and source terms for each affected module may be different.

3.4.4.4.2 Insights

The current regulatory framework for multi-unit and multi-module accidents does not include core damage or large release success criteria categorized by the number of failed units or modules. As such, a qualitative assessment of all potential multi-module effects from accident sequences that are screened in to the EPZ technical basis is required to provide a justification for severe accident selection while conforming to regulatory guidance.

3.4.5 Final Classification of Accidents by Severity

The final step of the methodology is to classify screened-in PRA accident sequences by containment integrity. Appendix I.B.2 of NUREG-0396 separates beyond-design-basis accidents into two broad consequence categories: less severe accidents and more severe accidents. Applicants demonstrate acceptability by classifying sequences as follows:

- intact containment sequences are less severe
- failed containment sequences are more severe
- if containment integrity for a given sequence is uncertain, consider both the less severe and more severe versions of the accident sequence

This classification of severity is based on the level of atmospheric release from containment, where more severe accidents involve a significant atmospheric release of radionuclides to the environment due to a compromised containment, and less severe accidents involve much smaller atmospheric releases due to an intact containment barrier. A review of NUREG-0396 by the NRC staff (Reference 6.1.11, Section 2.1) concluded that NUREG-0396 classified less severe accidents as accidents that did not involve containment failure and release to the atmosphere, while more severe accidents involve containment failure and large releases to the atmosphere. This methodology uses the same approach to accident classification as the past plume exposure pathway EPZ distance determination NRC precedent. Additionally, advanced designs such as NuScale have been designed to reduce the possibility of containment failure, consistent with NRC policy expectations.

If the sequence does not include a loss of containment integrity (i.e., the core damage progression does not include failure of the containment function), the accident sequence is classified as "less severe" and the dose evaluation is performed using the methodology of Section 4.2.2. The effect of an intact containment is that the only potential radionuclide release to the environment is by containment leakage; therefore, an intact containment is equated with "less severe."

If the methodology results in the screening out of every less severe accident sequence, no additional less severe surrogate accident sequence is required to be retained for source term and dose evaluation. This is because the requirement to include the DBST is by definition already a mandated analysis of a less severe accident sequence.

In sequences where containment integrity is not maintained for at least one module involved in an accident (i.e., containment bypass loss-of-coolant accident, containment isolation fails, or containment is otherwise breached), the accident sequence is classified as "more severe" and the dose evaluation is performed using the methodology of Section 4.2.3.

If the methodology results in the screening out of every more severe accident sequence, there is no requirement to include a more severe source term and dose evaluation.

If there is uncertainty as to the integrity of containment, the accident sequence should be considered both less and more severe, and evaluated against both dose criteria according to the following methods.

- Evaluate the intact containment version of the sequence against the 1 rem mean and 5 rem 95th percentile TEDE dose criterion (Section 4.2.2)
- Evaluate the containment bypass version of the sequence against the 200 rem acute red marrow {{

}}2(a),(c)

  - Perform a mechanistic analysis of the event and determine the relative frequency of containment failure conditional on the accident occurring, or
  - Conservatively use the frequency of both the intact containment and containment bypass versions of the accident sequence.

{{

}}2(a),(c)

3.5 Other Risks

Other risks with impact to EPZ sizing outside of the events identified in Section 3.3 and Section 3.4 may exist that are design-specific or site-specific, such as SFP risk, non-core damage events such as mechanical damage to the fuel following a postulated drop of the upper reactor module onto the lower reactor pressure vessel head in the reactor flange tool support during refueling, or risk from local volcanic activity. These other risks that may lead to potential off-site radionuclide releases must be identified and evaluated to ensure an appropriate planning distance. Applicants demonstrate acceptable consideration of such risks by meeting one of the following criteria:

- Dose-based criteria in Section 3.2, for the appropriate accident severity as described in Section 3.4.5
- Demonstration, qualitatively or quantitatively, that the consequences of each risk are bounded by events screened-in by Sections 3.3 and 3.4

This report provides acceptance criteria for two such risks, using the second criterion above: SFPs, and severe accident phenomena. Applicants with other risks in addition to the SFPs and severe accident phenomena demonstrated in the following sections shall propose acceptance criteria and demonstrate acceptability for those risks.

3.5.1 Spent Fuel Pool

Accidents involving the SFP may be eliminated from detailed consideration in the EPZ technical basis, provided the following three criteria are met:

- The time required to boil off SFP inventory is sufficiently long (i.e. >10 hours) that mitigating measures can be implemented to prevent fuel damage.

- Criticality is precluded during refueling operations and storage of spent fuel at maximum capacity, including following design-basis events.
- Leaks from the SFP can be detected and mitigated through monitoring and replenishment systems.

Sections 3.5.1.1, 3.5.1.2, and 3.5.1.3 provide examples based on the NuScale design of a demonstration of meeting the three acceptance criteria for SFPs.

3.5.1.1 Spent Fuel Pool Boil-Off

Under this scenario, all 12 reactor modules are simultaneously shut down with no active cooling of the UHS. The decay heat from the reactor modules and the maximum capacity of the SFP are the heat source. The UHS heats up and eventually evaporates with the conservative assumption that no water returns to the UHS from condensation in the Reactor Building. Once the UHS level drops below the elevation of the weir separating the spent fuel from the balance of the UHS pools, the spent fuel is the only heat source in the SFP, which continues to evaporate down to the top of the spent fuel.

Bounding calculations using simplifying and conservative assumptions are used to solve heat balance equations and determine time to UHS heatup and boiling. These calculations indicate that it takes several months to boil down the water to the top of the weir with additional time necessary for the water to boil down to reach the top of the spent fuel racks.

These results demonstrate that the NuScale plant design provides sufficient time to take mitigating measures such as replenishing the SFP water inventory, thereby preventing fuel damage.

3.5.1.2 Criticality

The NuScale methodology for criticality analysis is applied to the fuel assemblies stored in the SFP. This analysis accounts for the spent fuel fissile material (U and Pu), moderation, and geometry (storage and stacking arrangement).

The boron concentration in the SFP is maintained at a value that will preclude criticality during refueling operations. In addition, the Seismic Category I spent fuel storage rack includes poison panels that are independently capable of preventing criticality with no credit for boron concentration.

3.5.1.3 Leak from Spent Fuel Pool

Unlikely leaks from the SFP will be detected through the SFP sump liquid monitoring system. The SFP water makeup system will replenish water losses from unlikely leaks and evaporation.

In case of leaks occurring over extended periods of time, water supply from the balance of the UHS pools will supply coolant for fuel in the spent fuel storage racks to preclude uncovering them and thus prevent fuel damage.

In addition, an external source of water will be available to replenish the water inventory in the SFP. Moreover, the ability to provide makeup water to the SFP with a Seismic Category I connection outside of the Reactor Building has been included in the NuScale design.

3.5.2 Severe Accident Phenomena

While it is fully expected that severe accident phenomena will be analyzed as part of the underlying PRA used as input to the EPZ sizing method, nevertheless the potential impacts of severe accident phenomena on containment integrity must be accounted for. As discussed in Section 3.4.5, if containment integrity is not maintained for any reason, including occurrence of severe accident phenomena, the accident sequence is classified as "more severe." Applicants demonstrate acceptable consideration of potential severe accident phenomena occurring during screened-in accident sequences, provided one of the following criteria are met:

- Severe accident phenomena do not occur, or
- Containment integrity is maintained, or
- Loss of containment integrity due to severe accident phenomena does not significantly alter the source term to the environment

As an example using the NuScale design, relevant severe accident phenomena, including in-vessel retention, fuel-coolant interaction, hydrogen combustion in containment, and high pressure melt ejection have been assessed as part of the DCA. Current analyses predict that these severe accident phenomena either do not occur, or do not challenge containment integrity. However, the application of the EPZ methodology will consider the assessment of severe accident phenomena for a design at that time of a specific COL application.

3.6 Security Events

Security events are addressed for completeness for EPZ; however, accidents resulting from security events may be eliminated from detailed consideration in the EPZ technical basis. Applicants demonstrate acceptability by meeting regulatory requirements to protect against both design-basis and beyond-design-basis threats.

3.6.1 Design-Basis Threats

Applicants justify protection against design-basis threats by meeting regulatory requirements and describing security-by-design features of the plant.

For example, the NuScale design reduces the number of safety systems, thereby reducing the number of potential targets. For the remaining safety systems, most of the safety-related components have been located below grade. As the safety systems are passive in design, there is no reliance on operator actions, electrical power, or the addition of water to maintain the safety of the reactor cores or spent fuel.

A COL applicant will develop a site-specific strategy based on design-specific security features to protect against radiological sabotage, as outlined in Purpose and Scope, 10 CFR 73.1 (Reference 6.3.16).

3.6.2 Beyond-Design-Basis Security Events

Applicants demonstrate protection against beyond-design-basis events by meeting the requirements described in the following sections regarding aircraft impact and loss-of-large-area (LOLA) events.

3.6.2.1 Aircraft Impact

All new plants to be built and operated in the United States must meet the regulatory requirements for aircraft impact in Aircraft Impact Assessment, 10 CFR 50.150 (Reference 6.3.15). These regulations require that all new plant applications must:

1. Perform a design-specific assessment of the effects on the nuclear power plant facility of the impact of a large, commercial aircraft; and
2. Using realistic analyses, the applicant shall identify and incorporate into the design those design features and functional capabilities to show that, with reduced use of operator actions:
    a. the reactor core remains cooled, or the containment remains intact; and
    b. spent fuel cooling or SFP integrity is maintained.

For example, for DCA NuScale has performed structural and heat removal analyses, which substantiate that the Reactor Building meets the aforementioned regulatory requirements. The NuScale DCA Reactor Building is an aircraft impact resistant structure.

3.6.2.2 Loss of Large Area

Conditions of Licenses, 10 CFR 50.54, Section (hh)(2) (Reference 6.3.11), requires that new plants constructed and operated in the United States consider the LOLA of the plant due to fire or explosion. The regulation requires each licensee to develop and implement guidance and strategies intended to maintain or restore core cooling, containment integrity, and SFP cooling capabilities under the circumstances associated with LOLAs of the plant due to explosions or fire, and to include strategies in the following areas:

1. Firefighting
2. Operations to mitigate fuel damage
3. Actions to minimize radiological release

Applicants demonstrate compliance with LOLA regulatory requirements by performing an analysis for LOLA using the guidance provided in B.5.b Phase 2 & 3 Submittal Guidance, NEI 06-12 (Reference 6.3.13).

3.7 Defense-in-Depth Evaluation

Applicants demonstrate acceptability by completing a plant-level, qualitative evaluation of defense-in-depth as required in an application of the EPZ sizing methodology. Applicants evaluate defense-in-depth using the guidance in INSAG-10 (Reference 6.3.14) and RG 1.174 (Reference 6.3.2).

While independent and separate from the screening process, the evaluation is performed to confirm that the design and operation maintain consistency with the defense-in-depth philosophy. Defense-in-depth is an approach to design and operation that prevents and mitigates accidents. The key is providing programmatic controls and plant capability that create multiple independent and redundant layers of defense to compensate for potential human and mechanical failures so that no single layer, no matter how robust, is exclusively relied upon. Defense-in-depth is a way to account for uncertainties in equipment and human performance, and account for the potential for unknown and unforeseen failure mechanisms or phenomena.

The evaluation should address the five levels of defense identified by the International Atomic Energy Agency in INSAG-10 (Reference 6.3.14):

- prevention of abnormal operation and failures
- control of abnormal operation and detection of failures
- control of accidents within the design basis
- control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents
- mitigation of radiological consequences of significant releases of radioactive materials

It should also address the seven evaluation factors identified in RG 1.174 (Reference 6.3.2):

- preserve a reasonable balance among the layers of defense
- preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures
- preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty
- preserve adequate defense against potential common cause failures
- maintain multiple fission product barriers
- preserve sufficient defense against human errors
- continue to meet the intent of the plant's design criteria

A major objective of the defense-in-depth assessment is to highlight unique design features and SSCs available to mitigate the consequences of postulated accidents (e.g., Seismic Class I reactor building). While screening confirms the spectrum of accident sequences, identifying SSCs that serve as a physical or functional barrier to the transport of radionuclides will increase confidence in public protection. Mitigation strategies and SSCs should emphasize inherent improvements in safety and security and demonstrate protection of the public health and safety in the unlikely event of a release.

The defense-in-depth assessment will identify key plant characteristics that result in significant enhancements in safety (e.g., passive safety features, smaller cores, slower accidents). Consideration of severe accident management strategies and diverse and flexible coping strategies addresses mitigation of beyond-design-basis events. The assessment also confirms the existence, functionality, and capability of features and strategies to provide confidence in the acceptably low plant risk and demonstrate protection of the health and safety of the public; a defense-in-depth assessment is consistent with NRC's safety philosophy to show additional capability to address uncertainties.

3.8 Reviewing Key Uncertainties and Sources of Uncertainty in the Underlying PRA

Applicants demonstrate acceptability by completing a review of the assumptions and sources of uncertainty in the underlying PRA to identify and address any potential impact on the application of the EPZ sizing method.

As stated in Section 2.5, a prerequisite for using the NuScale EPZ sizing method is to use a full-scope, all-hazards PRA that has been established as technically acceptable. Therefore, only those uncertainty issues that can be directly related to sizing of an EPZ are relevant here. These issues include:

- Key assumptions in the PRA
- Model uncertainty
- Completeness uncertainty

4.0 Methodology for Source Term and Dose Evaluations

4.1 Application of Software

In evaluating the plume exposure EPZ size, severe accident and off-site consequence software will reflect the design, source terms, and severe accident dose characteristics. Applicants demonstrate acceptable use of software by meeting the following criteria:

- Calculate time-dependent source terms to the environment that reflect the spectrum of accident sequences identified in Section 3.4.
- Calculate site-specific off-site dose consequences for the time-dependent source terms to the environment.

The MELCOR and RELAP5 codes are recommended for source term calculations in this methodology. Use of any different computer codes should be technically justified and cover the same range of phenomena. MACCS, as an industry standard code that has been NRC-developed for dose calculations, is required for off-site dose consequence calculations in this methodology. For all codes used, it is recommended that the latest final and approved version released to all users by the developer be used. If another approved version is released during the analyses, it should be confirmed that the EPZ technical basis does not change with the newer version.

The RELAP5 code is a system thermal hydraulics code for analysis of transients and accidents up to the time of core damage. MELCOR is a fully-integrated thermal-hydraulic computer code that models the progression of severe accidents for a wide range of severe accident phenomena. In this methodology, the severe accident code (MELCOR) is the primary software used to calculate source terms. The user of the methodology will have confidence in the severe accident results, and one way to accomplish this is to use another thermal hydraulic code (e.g., RELAP5) to confirm accident progression up to core damage.

The MACCS code is used to calculate off-site dose consequences using MELCOR calculated time-dependent source terms and site-specific meteorology data.

The following sections generally assume that a user of this methodology individually evaluates each radionuclide release identified in Section 3.4 for dose consequences. Release categories, representing multiple sequences with similar accident progressions and release characteristics, may be used to reduce the number of severe accident and dose consequence evaluations while ensuring the frequency and consequence of each sequence is represented. The applicant must identify and justify the technical basis for the grouping of sequences into release categories, if used.

Upon request, a COL applicant should provide input and output files for each code used to the NRC to facilitate the review. The following subsections describe the computer codes noted above and general modeling requirements for each code.

4.1.1 Severe Accident Software and Modeling

There is not a specific severe accident software that must be used to comply with the EPZ sizing methodology. Applicants shall demonstrate the ability to evaluate the following severe accident phenomena:

- the transient thermal hydraulic response of the fuel and primary and secondary coolant systems
- fuel degradation and relocation
- fission product release and transport behavior
- challenges to barriers between fission products and the environment
- secondary chemical reactions (e.g., hydrogen production, transport, and combustion)

The user of the methodology needs to have reasonable confidence in the severe accident results. Confirmation of results up to core damage with another thermal hydraulic code is one option to provide confidence in the severe accident results.

To perform the accident source term evaluation, an integrated severe accident model shall be developed. A model may already be available from the site-specific PRA. The severe accident model shall contain, at a minimum:

- the primary coolant system,
- containment,
- important safety and nonsafety-systems, and
- associated control logic.

Severe accident analyses evaluate the selected sequences (or release categories, if used) to determine fuel/cladding failure with their concomitant radionuclide release fractions and timing. Fission product release is determined based on the amount of fuel damage resulting from the specific accident sequences determined to be part of the EPZ basis in Section 3.4.

Unique design features may exist that can mitigate the release of radioactivity to the environment. Examples of such features are a secondary containment or confinement, a reactor building, building sprays, external pools, and deposition surfaces in piping prior to the location of a pipe break. EPZ evaluations may credit mitigating design features; therefore, a severe accident code should have this modeling capability.

Separate-effects models may be employed to increase the fidelity of the severe accident simulations while simultaneously decreasing the computational burden of a larger integral model. For example, separate-effects containment bypass piping models may be developed for unisolated outside containment pipe break simulations. The piping may have significant internal surface area for deposition of fission product aerosols released from the fuel.

An example of the primary code to determine source terms is the MELCOR severe accident code, which is the reference example used throughout this report. MELCOR is a fully-integrated, engineering-level computer code that models the progression of severe accidents in LWR nuclear power plants. MELCOR is developed at Sandia National Laboratories for the NRC and models a broad spectrum of severe accident phenomena (MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide, NUREG/CR-6119 [Reference 6.4.1] and MELCOR Computer Code Manuals, Vol. 2: Reference Manual, NUREG/CR-6119 [Reference 6.4.2]). These include thermal-hydraulic response in the reactor coolant system, containment, and confinement buildings; core heat up, degradation, and relocation; hydrogen production, transport, and combustion; fission product release and transport behavior. The MELCOR code has been assessed against numerous separate-effects tests, integral tests, and actual accident studies by Sandia National Laboratories and other code users as discussed in Section 4 of MELCOR Best Practices as Applied to the SOARCA Project, NUREG/CR-7008 (Reference 6.4.11).

An example of a code that may optionally be used to confirm thermal hydraulic results in the primary code up to core damage is the RELAP5 code, which is the reference example used throughout this report. RELAP5 is a system transient thermal hydraulics code developed by Idaho National Laboratory (RELAP5/MOD3 Code Manual, NUREG/CR-5535, Reference 6.4.17). RELAP5 can only be used to model events up to the time of core damage because it lacks models for post core damage behavior. A model should be developed in NRELAP5 (or another code) for the purpose of analyzing the primary and secondary coolant system transients. The model should be a best estimate model and can serve as a basis for analyzing the system thermal hydraulic response of the plant for confidence in the primary severe accident code.

4.1.2 Off-site Consequence Software and Modeling

Off-site dose consequences are an integral part of determining an appropriate EPZ distance. Applicants demonstrate acceptability by meeting the following criteria:

- Use of the MACCS code for off-site dose consequence calculations.
- Modeling a stationary population without credit for protective actions, as described in Sections 4.2.1, 4.2.2, and 4.2.3.
- Use of a year of meteorological data representative of the site analyzed, selected from a minimum of 3 years of available meteorological data.
- Use of the appropriate dose conversion factors (DCFs) for TEDE and acute red marrow dose in MACCS.
- Use of an atmospheric transport and dispersion (ATD) model accepted by the NRC for use in the presence of building wake effects (i.e., the Ramsdell and Fosmire building wake model in MACCS version 4.1 or later), or confirmation of ATD results within 0.31 mi (0.5 km), as described in Section 4.2.4.

The MACCS code is required in this methodology to perform off-site consequence analyses. MACCS is developed by Sandia National Laboratories to simulate the impact of severe accidents at nuclear power plants on the surrounding environment (Code Manual for MACCS2 User's Guide, NUREG/CR-6613 [Reference 6.4.3]). MACCS is used to calculate the radiological release atmospheric transport and environmental dispersion. The principal phenomena considered in MACCS are radionuclide atmospheric transport and dispersion using a straight-line Gaussian plume model and plume depletion during downwind transport through radioactive decay, dry deposition, and wet deposition in the environment. The plume concentration and radionuclide deposition at a given distance from the radionuclide release are used to estimate short-term and long-term dose accumulation through several pathways important to the determination of a plume exposure EPZ including cloudshine, groundshine, inhalation, and deposition onto the skin.

The input to MACCS will describe the source term for each screened in sequence (or release category, if used) generally following the guidance in MACCS Best Practices as Applied in the SOARCA Project NUREG/CR-7009 (Reference 6.4.4). A dose-in-place model will be created where the population stays in place with no sheltering, relocation, or evacuation. The exposure durations used to determine dose are dependent on sequence classification. Specific duration times are included in Sections 4.2.1, 4.2.2, and 4.2.3.

MELMACCS is a processing tool that is used to transform MELCOR output data from a plot file into MACCS input. If MELCOR is used to produce the source terms for dose calculations, it is recommended that MELMACCS be used to calculate both deposition velocity (using the expert elicitation and gravitational settling hybrid option) and segment durations (by subdividing the radionuclide release into plume sections with distinct durations). Segments should match the smallest available resolution of weather data to take full advantage of wind shift, unless the RG 1.145 plume meander model is employed. The RG 1.145 plume meander model is specifically designed for durations of 1 hour. Specific parameter recommendations to determine deposition velocity are MELMACCS version-specific, but should follow the MACCS best practices document (Reference 6.4.4). If MAAP (Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using MAAP5 Software-Phase 3 Report [Reference 6.4.12]) or another computer code is used, the same methodology to segment the release and estimate deposition velocities should be employed without using MELMACCS.

A meteorological file should be created by obtaining meteorological data available for five years that is most representative of the meteorological conditions at the site, preferably at a location close to the site, and performing EPZ analyses separately for each year. These meteorological files should represent the highest resolution data available that can be utilized by MACCS (i.e., 15 minute average intervals and 64 azimuthal directions). The five years of data need not be consecutive. If five years of data are not available, a minimum of three years of data may be used. If three years of data are used, a statistical analysis of the distribution of stability classes for each of these three years should be performed to demonstrate that an adequate sample of site-expected meteorological data has been utilized. The SOARCA method (Reference 6.2.14) should be used to fill in missing meteorological data assuming there is not 100-percent recovery from meteorology measurement instrumentation.

The year that results in the largest dose for each acceptance criteria will be used in the final analyses. Meteorological data should be sampled in a stratified random manner, taking readings from the file every hour over the entire year as the starting point for the release.

The *DOSD60.inp* DCF file included with the legacy version 1.13.1 of MACCS should be used for TEDE dose as recommended with caveats by Use of MACCS Dose Coefficient Files to Compute Total Effective Dose Equivalent (Reference 6.4.18). The *DOSD60.inp* files utilize the appropriate tissue weighting factors for International Commission on Radiation Protection publications 26 and 30, as codified in 10 CFR 20.1003. The most recent DCF file included with the latest version of MACCS should be used for acute red marrow dose. These DCF files should either be unmodified, or any modifications should be technically justified.

The radial distance intervals should start at 0.031 mi (0.05 km) and extend out to at least 10 mi (16.1 km). Beginning with version 4.1, MACCS implements the Ramsdell and Fosmire building wake and plume meander model (Implementation of Additional Models into the MACCS Code for Nearfield Consequence Analysis, Reference 6.2.19) that has been accepted by the NRC for plume dispersion and transport in the presence of building wakes. No further confirmation of results is necessary if an NRC approved model for plume dispersion in the presence of building wakes (e.g., the Ramsdell and Fosmire model) is used to estimate dose consequences at distances within 0.31 mi (0.5 km) from the source.

Due to industrial-scale building wake effects, estimated doses at distances less than 0.31 mi (0.5 km) from the source are subject to uncertainty in the area source building wake model employed in legacy versions of MACCS. No attempt to quantify this uncertainty is documented in the MACCS manual. Department of Energy (DOE) guidelines for the use of MACCS conclude that MACCS dose versus distance results should be carefully checked within 0.31 mi (0.5 km) (MACCS2 Computer Code Application Guidance for Documented Safety Analysis, Final Report, DOE-EH-4.2.1.4 [Reference 6.4.6]). The DOE guidelines stress the use of caution to ensure accurate results are reported in the first several hundred meters of plume travel, as the plume concentration in this region may be highly influenced by plume buoyancy and other near field dispersion phenomena. Additional steps to ensure validity of results from the area source building wake model, if used, at distances less than 0.31 mi (0.5 km) are discussed in Section 4.2.4.

To provide additional confidence in the MACCS code, a code comparison was performed in Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model, NUREG/CR-6853 (Reference 6.4.7) to benchmark MACCS against other ATD codes [i.e., NRC's codes RASCAL (Reference 6.4.13) and RATCHET (Reference 6.4.14), and Lawrence Livermore National Laboratory's computer code ADAPT/LODI (Reference 6.4.15)]. RASCAL is the NRC's computer code for rapid emergency response; whereas, RATCHET is a newer related code with upgraded dispersion and deposition modules. ADAPT/LODI is a state-of-the-art, three-dimensional, advection dispersion code. Agreement among the models used in NUREG/CR-6853 was considered acceptable.

Although the distances at which the codes were compared in NUREG/CR-6853 extend beyond the region of interest for EPZ sizing (i.e., >10 miles or 16 kilometers), the results provide confidence in the validity of the MACCS code.

4.2 Source Term and Dose Evaluation Methodology

This section discusses the methodology for performing the evaluation of the source term and dose for the accident sequences determined to be appropriate as part of the EPZ size basis in accordance with Section 3.0. Applicants demonstrate acceptability by meeting the following criteria:

- Calculation of source dose consequences for the FSAR Chapter 15 off-site DBST and comparison to Dose Criterion a.
- Calculation of source terms and dose consequences for each screened-in less severe accident sequence and comparison to Dose Criterion b.
- Calculation of source terms and dose consequences for each screened-in more severe accident sequence and comparison to Dose Criterion c.

To aid in overall understanding of Section 4.0, Table 4-1 provides a summary of the methodology. For the three accident types to be addressed (DBA, less severe accidents, and more severe accidents), Table 4-1 includes the major steps involved in developing the EPZ size basis (i.e., source term evaluation and dose evaluation).

Table 4-1 High-level summary of the source term and dose methodology for developing EPZ size basis

<table>
<tr><th>Accident Type</th><th>Source Term Evaluation</th><th>Dose Evaluation</th></tr>
<tr><td>DBA
(Section 3.3)</td><td>Required:
• Apply DBST fission product release from containment for off-site dose calculated for FSAR Chapter 15
• No credit for additional mitigating features beyond containment</td><td>• Apply MACCS dose-in-place evaluation (different from Chapter 15 dose evaluation)
• Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used
• Dose Criterion a (Section 3.2) with exposure duration of 96 hours</td></tr>
<tr><td>Less severe accidents
(Section 3.4)</td><td>Required
• Apply full module severe accident model to calculate fission product release from the module due to containment leakage

Optional:
• Apply additional mitigating design features to calculate fission product holdup and deposition, and fission product release to environment
• Consider all operator mitigation actions</td><td>• Apply MACCS dose-in-place evaluation
• Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used
• Dose Criterion b (Section 3.2) with exposure duration of 96 hours</td></tr>
</table>

Table 4-1 High-level summary of the source term and dose methodology for developing EPZ size basis (Continued)

Accident Type	Source Term Evaluation	Dose Evaluation
More severe accidents (Section 3.4)	**Required** • Apply full module severe accident model to calculate fission product release from the module through containment bypass piping **Optional:** • Apply separate effects piping models to calculate fission product deposition in containment penetrations, credit other mitigation features such as a reactor building, pool, and building sprays, and calculate fission product release to the environment • Consider all operator mitigation actions	• Apply MACCS dose-in-place evaluation • Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used • Dose Criterion c (Section 3.2) with exposure duration of 24 hours

4.2.1 Design-Basis Accidents

The first accident category evaluated to determine the EPZ distance is the off-site DBST. Applicants demonstrate acceptability by meeting the following criteria:

- Compare the off-site DBST against Dose Criterion a considering:
  - site-specific meteorology and shielding factors
  - cloudshine, groundshine, inhalation, and resuspension dose pathways
  - no protective actions
  - 96 hour exposure duration following plume arrival
  - EFFECTIVE DCFs
  - no credit for structures external to containment

<u>Source Term Evaluation Methodology</u>

The accident sequence source term evaluation methodology for Dose Criterion a (1 rem mean TEDE and 5 rem 95th percentile TEDE, see Section 3.2), is to utilize the DBST, which is a time-dependent fission product release from containment to the environment that will be used to analyze off-site dose as part of the COL application and can be extracted from Chapter 15 of the FSAR. Thus, little or no additional work is expected to be necessary for application of the methodology for either determining the appropriate DBA to be evaluated (see Section 3.3) or for the source term evaluation in conjunction with implementing Dose Criterion a. The release of fission products from containment is based on the assumption of design-basis leakage at containment design pressure, and the assumption that containment leakage occurs directly to the environment.

Dose Evaluation Methodology

The EPZ dose evaluation for the DBA is based on Section 3.3 of the NEI white paper (Reference 6.1.5). A summary of the methodology to be used for the dose evaluation is contained within this section. It should be noted that the EPZ dose calculation methodology for Dose Criterion a differs from that used for off-site dose in Chapter 15 analyses in that it is based on the methodology typically used in severe accident dose calculations; in particular, use of the MACCS software. The methodology includes the following elements:

- The EPZ boundary dose calculation will apply a methodology similar to that used in the SOARCA study (Reference 6.2.14), which used MACCS state-of-the-art consequence analysis software.
- MACCS input parameters for the applicable site and design-specific source terms will be developed. An example of a site-specific MACCS model can be found in the Surry SOARCA study (Reference 6.4.5).
- Dose Criterion a will be applied. The PAG dose values are 1 to 5 rem TEDE (Reference 6.1.4) projected for an exposure duration of 96 hours. The mean dose at the EPZ boundary will be less than 1 rem TEDE and the 95th percentile dose at the EPZ boundary will be less than 5 rem TEDE. Mean and 95th percentile doses are calculated by MACCS and are based on statistical sampling and evaluation of the meteorological site data that will be created as an input to MACCS.
- TEDE will be calculated for cloud, inhalation, ground, and resuspension. The current recommended MACCS output parameter is the EFFECTIVE organ found in the *DOSD60.inp* DCF file (Reference 6.4.18).
- When the wind shift without rotation plume model is used, which is the current best practice, peak dose[5] on the spatial grid is the desired output (Reference 6.4.3, Section 6.19). If a plume model without wind shift is used, peak centerline dose is the desired output.
- Meteorological files with 64 azimuthal sectors should be used. A new meteorological file should be created as discussed in Section 4.1.2.
- Stratified random sampling of the meteorological data shall be used to access the meteorological file every hour over the entire year as the starting point for the release.
- An NRC-accepted model for plume dispersion in the presence of building wakes is applied (e.g., the Ramsdell and Fosmire model).
- If the legacy area source building wake model is used, MACCS is applied for distances down to 0.5 km (0.31 miles) from the reactor. For smaller distances, additional steps discussed in Section 4.2.4 are used to address concerns with area source building wake model MACCS results.

5. It is important to note that "peak dose" is used in multiple contexts in MACCS output. "Peak dose" can refer to the maximum dose on the spatial grid for a given set of meteorological conditions, or the maximum dose over all weather trials considered. Here, "peak dose" refers to the maximum dose that occurs on the spatial grid.

- In performing the dose evaluation, the EPZ will encompass those areas in which projected dose from DBAs could exceed the PAGs.
- Site-specific shielding and protection factors can be used with technical justification; otherwise, the conservative values from NUREG-0396 (Reference 6.1.3) (0.7 for groundshine dose, and 1.0 [no shielding] for cloudshine dose and 1.0 [no protection factor] for inhalation dose) should be used.
- Initially, a dose-in-place model will be used with no ad hoc protective actions taken. As a second step, ad hoc, site-specific protective actions may be considered to determine impact on public dose, including relocating people from regions outside the EPZ, after initial calculations without protective actions determine the EPZ distance. When considered, the COL applicant shall apply a site-specific normal and hotspot relocation time, with appropriate justification.

Parameters and the recommended default values to ensure a stationary population and EFFECTIVE dose exposure for the initial 96 hours (345,600 seconds) following plume arrival are shown in Table 4-2. Shielding values (and relocation times for the optional confirmatory calculation) may be updated to reflect site-specific information.

Table 4-2 MACCS parameters and recommended default input values for dose-in-place model to evaluate Dose Criteria a and b

MACCS Input	Parameter	Value
Dose Type	EFFECTIVE	N/A
Exposure Duration	ENDEMP	345600 s
Cloudshine Shielding	CSFACT	1.0
Groundshine Shielding	GSHFAC	0.7
Inhalation Shielding	PROTIN	1.0
Evacuation	EVATYP	NONE
Normal Relocation Time	TIMNRM	345600 s
Normal Relocation Dose	DOSNRM	1.0E+10 Sv
Hotspot Relocation Time	TIMHOT	345600 s
Hotspot Relocation Dose	DOSNRM	1.0E+10 Sv

4.2.2 Less Severe Accidents

The second accident category that may be evaluated to support EPZ distance is less severe accidents. Applicants demonstrate acceptability by meeting the following criteria:

- All accident sequences determined to be less severe accidents in Section 3.4.5 are compared against Dose Criterion b, considering:
  - site-specific meteorology and shielding factors
  - cloudshine, groundshine, inhalation, and resuspension dose pathways
  - no protective actions
  - 96 hour exposure duration following plume arrival

- EFFECTIVE DCFs

Section 3.4 addresses the methodology for determination of appropriate less severe accident sequences to be evaluated as part of the basis for EPZ size. For the appropriate less severe accident sequences, evaluations of the source term and dose are performed and compared to Dose Criterion b (1 and 5 rem TEDE). The source term evaluation will calculate the fission product release to the environment versus time, which is then used as the input to the dose evaluation.

The methodology for performing the source term and dose evaluations for less severe accidents is contained within this section.

Source Term Evaluation Methodology

The methodology for the source term evaluation will apply the severe accident analysis thermal hydraulic software, with MELCOR and RELAP5 used as examples. The methodology includes the following elements:

- Development and benchmarking of the primary severe accident (e.g., MELCOR) model as discussed in Section 4.1. This MELCOR module model is used to calculate radionuclide release fractions from containment, as a function of time, for the less severe accident sequences that were determined as appropriate for evaluation to support EPZ.
- The impact on source term of operationally-focused mitigation (i.e., SAMGs, EDMGs, and other EPZ-oriented operator mitigation actions in addition to EOPs) may be considered.
- The leakage of fission products from containment may be calculated by the severe accident code, and should be based on the conservative assumption of a small opening sized to result in design-basis leakage at containment design pressure (i.e., the technical specification limit). Alternatively, the fission product release to the environment from a containment modeled as perfectly sealed may be calculated by multiplying the airborne radionuclide concentration in containment as a function of time by the design-basis leakage rate at containment design pressure.
- A model of additional fission product barriers external to the containment may optionally be used to calculate fission product deposition and holdup following leakage from the containment. The source term to the environment is then used as input to the dose evaluation.

Dose Evaluation Methodology

The EPZ dose evaluation methodology for the less severe accident sequences is based on Section 3.3 of the NEI white paper and applies Dose Criterion b (Section 3.2). The methodology is the same as that for DBAs described in Section 4.2.1.

4.2.3 More Severe Accidents

The final accident category that may be evaluated to support EPZ distance is more severe accidents. Applicants demonstrate acceptability by meeting the following criteria:

- All accident sequences determined to be more severe accidents in Section 3.4.5 are compared against Dose Criterion c, considering:
  - site-specific meteorology and shielding factors
  - cloudshine, groundshine, inhalation, and resuspension dose pathways
  - no protective actions
  - 24 hour exposure duration following plume arrival
  - acute red marrow DCFs

Section 3.4 also addresses the methodology for determination of more severe appropriate accident sequences to be evaluated as part of the basis for EPZ size. For any more severe accident sequences that are screened in to the EPZ technical basis source term and dose evaluations will be performed in accordance with the methodology. Dose Criterion c {{

}}2(a),(c) is applied.

The methodology for performing the source term and dose evaluations for more severe accidents is contained within this section.

<u>Source Term Evaluation Methodology</u>

The methodology for the source term evaluation applies severe accident software in the same manner as for less severe accident sequences in Section 4.2.2, with the following exceptions:

- Fission product release occurs through containment bypass flow paths and, therefore, consideration of design-basis containment leakage is not required.
- In addition to the models of additional external barriers, a separate effects model should be used to credit fission product deposition in the release pathway, such as piping, between the containment and the additional external barriers.

<u>Dose Evaluation Methodology</u>

The EPZ dose evaluation methodology for more severe accident sequences is {{

}}2(a),(c) The evaluation of Dose Criterion c is consistent with the dose evaluation methodology in Section 4.2.1, with the following exceptions:

- Dose Criterion c is applied (i.e., the EPZ should be of sufficient size to provide for substantial reduction in early severe health effects in the event of more severe accidents).

- Consistent with NUREG-0396 (Reference 6.1.3) and the NEI white paper (Reference 6.1.5), dose will be calculated based on the following:
  - The metric to be used for "substantial reduction in early severe health effects" is 200 rem whole body acute dose. Red bone marrow (the A-RED MARR MACCS output parameter) is an acceptable surrogate for acute whole body dose (Reference 6.1.3, Section III, Subsection D). As discussed later in this section, a "substantial reduction in early severe health effects" is considered to occur {{

    }}2(a),(c)
  - The exposure pathways (with exposure time in parentheses) should be cloudshine (24 hours), inhalation (30 days), and groundshine (24 hours). The cloudshine and groundshine exposure durations are controlled by MACCS EARLY input parameters. The inhalation exposure duration is incorporated into the calculation of the DCFs and is not controlled by MACCS input parameters
    - the current default DCF file uses a duration of 30 days for acute red marrow exposure from inhalation, based on MELCOR Accident Consequence Code System (MACCS), Volume 2: Model Description, NUREG/CR-4691 (Reference 6.4.9, Table D-5) and WinMACCS, a MACCS2 Interface for Calculating Health and Economic Consequences from Accidental Release of Radioactive Materials into the Atmosphere: User's Guide and Reference Manual (Reference 6.4.10, Appendix C-8).

Parameters and the recommended default values to ensure a stationary population and A-RED MARR dose exposure for the initial 24 hours following plume arrival are shown in Table 4-3. Shielding values (and relocation times for the optional confirmatory calculation) may be updated to reflect site-specific information.

Table 4-3 MACCS parameters and recommended default input values for dose-in-place model to evaluate Dose Criterion c

MACCS Input	Parameter	Value
Dose Type	A-RED MARR	N/A
Exposure Duration	ENDEMP	86400 s
Cloudshine Shielding	CSFACT	1.0
Groundshine Shielding	GSHFAC	0.7
Inhalation Shielding	PROTIN	1.0
Evacuation	EVATYP	NONE
Normal Relocation Time	TIMNRM	86400 s
Normal Relocation Dose	DOSNRM	1.0E+10 Sv
Hotspot Relocation Time	TIMHOT	86400 s
Hotspot Relocation Dose	DOSNRM	1.0E+10 Sv

For more severe accident sequences that are screened-in to the EPZ basis as determined in Section 3.4, {{

}}$^{2(a),(c)}$ This methodology is described in more detail in the subsequent paragraphs. {{

}}$^{2(a),(c)}$

In equation form {{

}}$^{2(a),(c)}$

A simple numerical example illustrating the steps {{

}}$^{2(a),(c)}$ In this example, the EPZ distance would be approximately 0.03 miles, as shown in Figure 4-1.

Table 4-4 {{ }}2(a),(c)

{{

}}2(a),(c)

Figure 4-1 {{

}}2(a),(c)

{{

}}2(a),(c)

With MACCS, the recommended method for determining the conditional probability of dose exceedance versus distance for a given sequence (p_{ij}) is to use the complementary cumulative distribution function output option in the WinMACCS EARLY output control parameter list. When the complementary cumulative distribution function option is selected for a given dose output, a table is printed (for selected distances) that contains various dose values and the probability of exceeding those doses. The uncertainty in the meteorological conditions is captured in this output, as the probability of dose exceedance result generated by MACCS explicitly evaluates all weather trials and provides the fraction of weather trials during which a given dose is exceeded at a given distance. An example of input and output for 200 rem acute red marrow dose exceedance is presented in Figure 4-2 and Figure 4-3, respectively.

Figure 4-2 Example WinMACCS complementary cumulative distribution function input selection

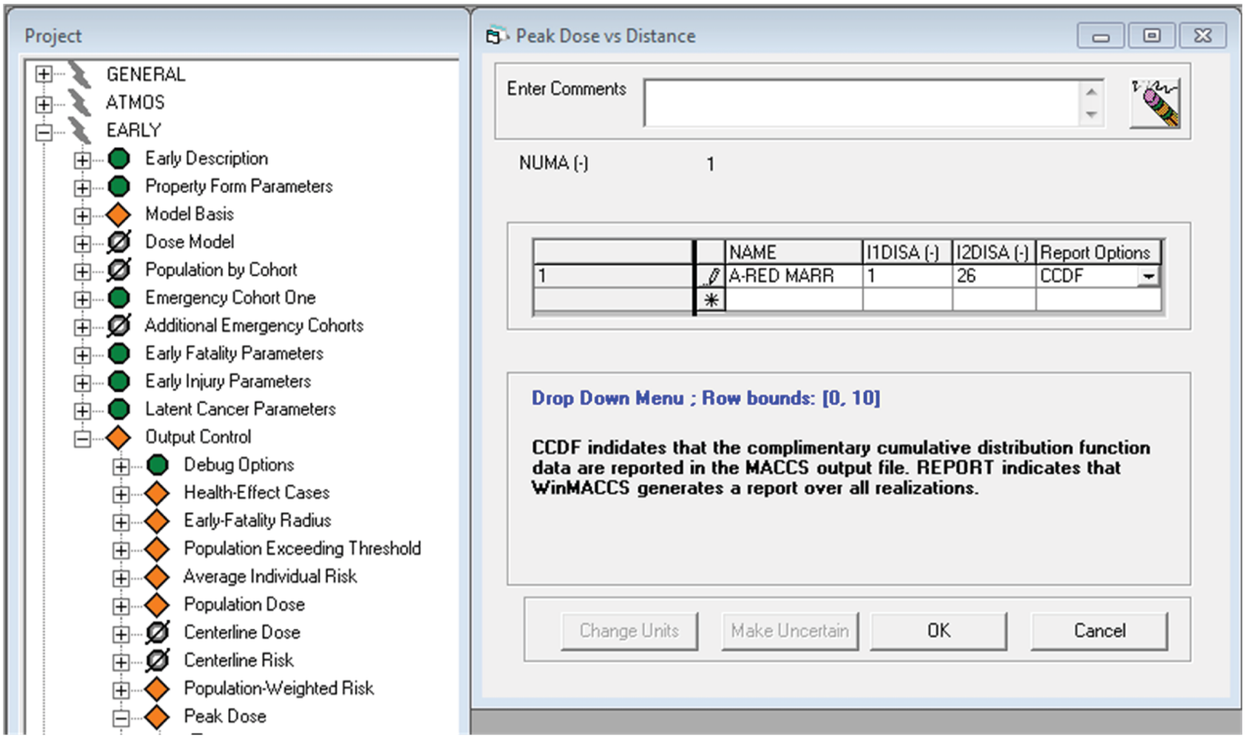


Figure 4-3 Example complementary cumulative distribution function output for a single distance

```
RESULT NAME = CENTERLINE DOSE AT SOME DISTANCES (rem)
                  A-RED MARR  TOT ACU         0-0.8 km

   PEOPLE FRACTION =      1.0000

  EMER. RESP. # 1
 -----------------
     X       PROB>=X
 1.00E+01  8.14E-01
 2.00E+01  6.43E-01
 3.00E+01  4.85E-01
 5.00E+01  3.59E-01
 7.00E+01  3.23E-01
 1.00E+02  2.87E-01
 2.00E+02  5.94E-02
 3.00E+02  3.71E-02
```

4.2.4 Additional Steps for Dose Evaluation Inside 0.5 Kilometers

MACCS near-field plume concentration calculations require validation within 0.5 km of the release when the legacy area source building wake model is implemented. The MACCS area source building wake model results can be used within 0.5 km if compared to an NRC accepted ATD code. Applicants demonstrate acceptability for an EPZ distance within 0.5 km by meeting one of the following criteria:

- Use of an NRC-accepted ATD model in MACCS for plume dispersion in the presence of building wakes (e.g., the Ramsdell and Fosmire model).
- The ratio of MACCS area source building wake model plume relative concentration to the NRC accepted ATD code calculated plume relative calculation is > 1.0 at the site boundary, the calculated EPZ distance, and 0.5 km.
  - It is acceptable to use changes in MACCS building wake modeling to achieve defensible (i.e. best-estimate) atmospheric dispersion values.

The MACCS user guide (Reference 6.4.3) cautions against the application of the MACCS area source building wake model for distances less than 0.5 km based on reference to field measurements and Gaussian model applicability in the wake of large buildings. To address this concern, the COL applicant will perform additional steps as part of the dose evaluation to confirm or improve the MACCS modeling inside 0.5 km.

A DOE-sponsored review of the MACCS code (Reference 6.4.6) discusses the implied restriction in the MACCS user guide (Reference 6.4.3) and effectively recommends a minimum applicability distance for MACCS of 0.1 km if the code is used with appropriate care. In this methodology, the "appropriate care" will be the confirmation and/or adjustment of MACCS modeling inside 0.5 km to address the building wake effect. This will be accomplished by comparing MACCS results with that from a computer code that has been previously accepted by the NRC for ATD modeling in the presence of building wakes, such as the NRC-sponsored ARCON96 (Atmospheric Relative Concentrations in Building Wakes, NUREG/CR-6331 [Reference 6.4.8]), which is used for smaller distances where building wake effects can be important. ARCON96, for example, is designed to provide dispersion results, which can be applied to control room dose (i.e., small source to receptor distances in the presence of building wake effects).

The main parameter to be used for the comparison is X/Q, which is an atmospheric dispersion numerical value. The building wake effect potentially impacts the airborne concentration for a given release rate at short distances from the trailing edge of the building. The four exposure paths in MACCS which could be important for EPZ dose are cloudshine, inhalation, groundshine, and resuspension. Dose for all of these exposure paths is calculated based on X/Q.

To compare the X/Q results, a ground level, zero plume energy release to the environment shall be modeled. The modeling of building wake effects in both codes should follow the respective modeling best practices for both codes. Directionally independent, ground level X/Q values shall be calculated at the minimum distance from the release point to the site boundary and 500 meters from the release source, for every hour of available yearly meteorological data. In both the ATD and MACCS codes, output options should be selected such that X/Q values are output for every weather trial and all calculated X/Q values can be compared.

The ratio of the MACCS X/Q to the NRC accepted code X/Q values for mean and 95th percentile X/Q, respectively, shall be compared at each distance up to 500 meters. These mean and 95th percentile X/Q ratios shall be greater than or equal to 1.0 at the minimum distance to the site boundary and 500 meters for the MACCS results to be considered acceptable within 500 meters. Following the determination of the EPZ distance, and only if the EPZ distance is determined to be less than 500 meters, the MACCS to NRC accepted code comparison shall be repeated at the EPZ distance to confirm that the mean and 95th percentile X/Q ratios remain greater than or equal to 1.0.

In the event that the MACCS X/Q values at distances inside 0.5 km do not meet the above criteria, MACCS building wake modeling changes will be incorporated and evaluated within 500 meters to ensure calculated doses are based on defensible atmospheric dispersion values. Examples of potential modeling changes include, but are not limited to, the use of a minimum value for the initial sigma-y and sigma-z, as discussed in the DOE-sponsored review of the MACCS code (Reference 6.4.6), use of the lookup table option for sigma-y and sigma-z, and use of the RG 1.145 plume meander model. The estimated EPZ distance is determined using a best estimate approach and, therefore, MACCS modeling changes may also be used to improve agreement in atmospheric dispersion results between the two codes in cases where the X/Q ratio is larger than 1.0.

4.3 Uncertainty Analysis Methodology

The purpose of the uncertainty analysis is to understand the important sources of uncertainty in the technical basis for EPZ size and to provide additional confidence in the best-estimate parameters used. Applicants demonstrate acceptability by following the methodology in Figure 4-4 with the following steps:

1. Selection of accident sequences (i.e., the most probable less severe and more severe accident sequences)
2. Identification of source term and dose calculation uncertainty parameters and definition of their distributions
3. Sampling of uncertainty parameters to generate sample inputs for MELCOR and MACCS calculations
4. Integrated Monte Carlo simulation using MELCOR and MACCS codes using sample inputs
5. Uncertainty analyses of the MELCOR and MACCS results, including identification of important parameters (i.e., parameters that contribute more than 5 percent of the total regression metric)
6. Confirmation of best-estimate values for important parameters
7. Repetition of source term and dose consequence analyses, if necessary

The uncertainty methodology provides reasonable assurance that uncertainty in the severe accident space is being addressed. Both epistemic and aleatory uncertainties are considered, primarily due to potential variations in input parameters used in the EPZ size evaluation. The strategy for addressing uncertainty in the EPZ sizing analysis is as follows:

- Base the EPZ size on detailed, best-estimate calculations of source term and dose results. This approach also provides more transparency in the results and a better basis for developing effective emergency plans.
- Apply the uncertainty analysis to strengthen confidence in the best-estimate results. This is to be accomplished with a mechanistic, state-of-the-art uncertainty analysis that considers the full range of uncertainty, as opposed to use of an arbitrary, generic confidence level, which can lead to excess conservatism and obscure useful information.

This strategy for addressing uncertainty is consistent with the assumptions in Section 3.1 for use of risk-informed methods in the EPZ sizing methodology. The EPZ methodology is an improvement over the 1970s use of a qualitative, generic concept for determining safety margin adequacy. Due to integrated uncertainty analysis in the methodology, it is unnecessary to resort to conservative solutions.

The framework for EPZ sizing uncertainty analysis is based on SOARCA Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station, NUREG/CR-7155 (Reference 6.4.16). For simplicity in describing the methodology, "MELCOR" is used to describe whichever severe accident code is used in EPZ analyses.

The selection of accident sequences to be addressed in the uncertainty analysis is Step 1. The selection process is to select the highest frequency less severe accident sequence and the highest frequency more severe accident sequence that were determined to be screened in to the EPZ technical basis according to the methodology in Section 3.4. If there are no severe accident sequences for which a source term and dose evaluation is required, then no uncertainty analyses of such accident sequences is necessary.

The sequence selection is then followed by the identification of uncertainty parameters for MELCOR and MACCS models that have a significant impact on the off-site radiological releases and the dose consequences. Both epistemic and aleatory parameters should be considered. The parameters from the SOARCA uncertainty analyses should be used as a starting point.

Once the uncertainty parameters are identified, their distributions are defined so that the sampling of the uncertainty parameters can be carried out to generate sample inputs for MELCOR and MACCS calculations. As appropriate, SOARCA distributions can be used for any parameters retained from that analysis. If any distributions are inappropriate for the analyzed design, or for any new parameters, distributions will be technically justified and, when possible, based on physical bounds. Along with the parameters varied, it is also recommended that a table be included with an application that details all parameters considered but ultimately rejected, along with the technical justification for non-inclusion.

For Steps 3 and 4, MELCOR is run with the sampled inputs resulting in a set of source terms or off-site radiological releases over 72 hours. Each source term is paired with a sampled input set in MACCS to produce a set of dose results.

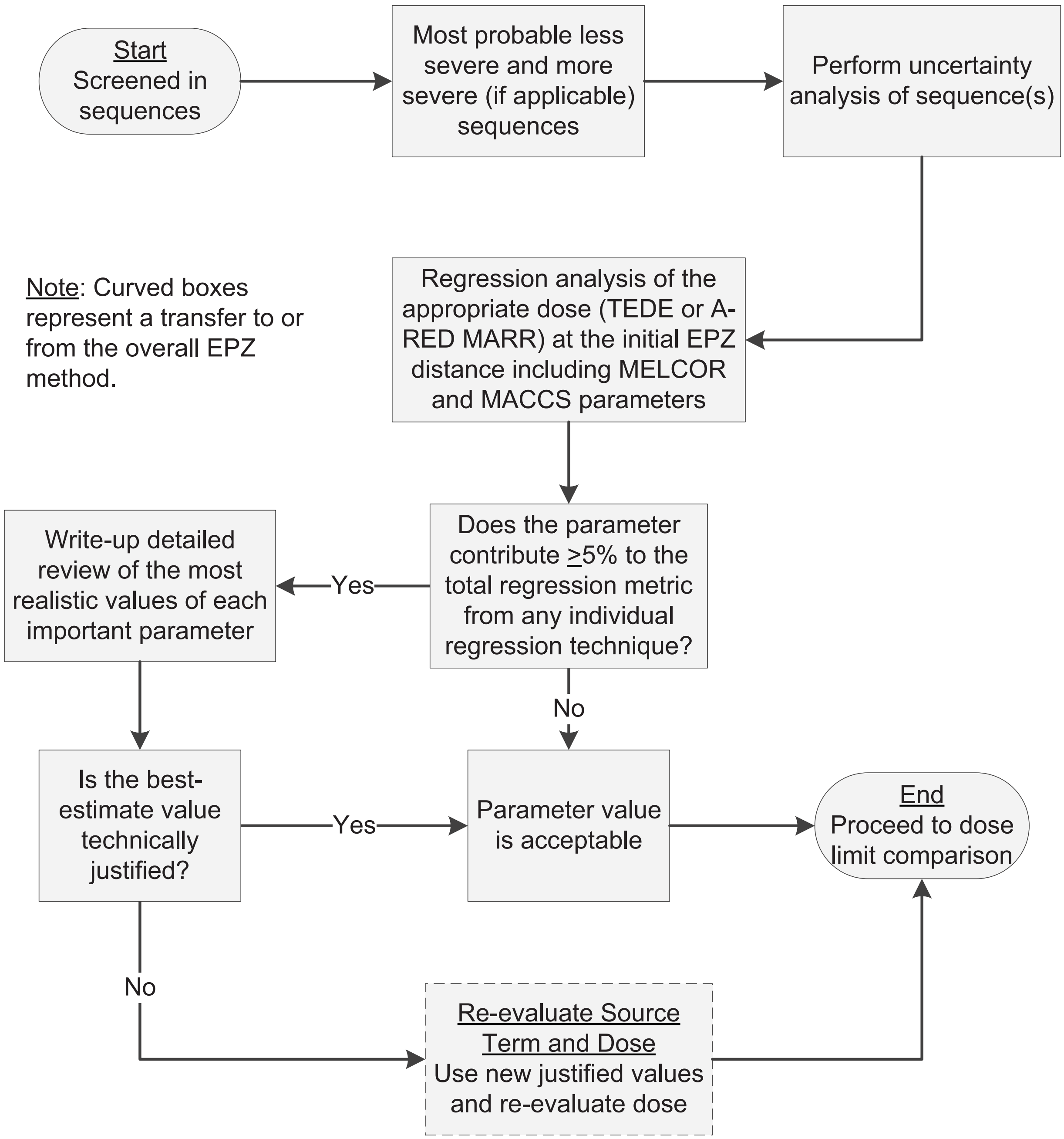
In Step 5, uncertainty analyses are performed on the results after all MELCOR and MACCS runs are completed. The uncertainty analysis is performed with one or more regression techniques (i.e., rank regression, quadratic regression, recursive partitioning, and multivariate adaptive regression splines). These analyses are performed on variabilities of the specified MELCOR and MACCS results that derive from uncertainties in the input parameters, and the contributions to the variabilities that derive from individual inputs.

The determination of important parameters and confirmation of best-estimate parameter values (Step 6) connects the uncertainty analysis methodology to the overall EPZ methodology. The metric used is the dose of the appropriate type (TEDE [EFFECTIVE in MACCS] for less severe, acute whole body [A-RED MARR in MACCS] for more severe) at the initially calculated EPZ distance. Using one or more of the regression techniques as described above, any MELCOR or MACCS parameter that is an important contributor to uncertainty is determined. An "important" contributor is defined as a parameter that contributes greater than 5 percent of the total regression metric. For example, the metric for rank regression is the coefficient of determination or R^2. If the total R^2 for rank regression is 0.60, any parameter with an individual R^2 of 0.03 or greater would be considered important. Five percent is chosen as the criteria because regression analysis tends to over predict correlation from random results, meaning that below 5 percent it becomes increasingly likely that parameters are identified because of the tool used as opposed to any real trend.

The important parameters require a dedicated write-up reviewing the source of the best-estimate value used. The best-estimate value will be technically justified based on experimental data, physical limits, and/or detailed design information. If the current best-estimate value is not justified, a new, justified value will be used and the MELCOR and MACCS analyses repeated (Step 7). A summary of the entire process is shown in graphical form in Figure 4-4.

The most important parameters cause the most variability in results, including the maximum and minimum outliers. Determination of the most important parameters considers the full range of uncertainty. By justifying the values used for the most important parameters additional confidence is given to the best-estimate analyses without the need to apply unnecessary conservatism.

Figure 4-4 Application method of uncertainty results



5.0 Summary and Conclusions on Methodology

The NuScale proposed approach for developing the technical basis for plume exposure EPZ size utilizes the 2013 NEI white paper framework and incorporates applicable concepts from the original, generic 1978 EPZ size basis in that it is dose-based and has a consequence orientation. At the same time, important differences exist in the NuScale approach including:

- it applies the severe accident knowledge base and analytical methods developed over the four decades since the original EPZ basis was formulated
- it is designed to be comprehensive, transparent, and repeatable

In addition, given the extent of PRA development and the evolution of risk-informed regulatory applications over the last several decades, NuScale is using risk-informed methods for determining appropriate accident sequences to be evaluated for the EPZ size basis. This risk-informed approach includes PRA information, deterministic source terms, and a qualitative evaluation of defense-in-depth.

This LTR submits a proposed plume exposure EPZ sizing methodology for NRC review. NuScale requests, as part of this review and associated comment resolution, that the NRC provide a safety evaluation report on the sizing methodology, including:

- a conclusion that the NuScale proposed plume exposure EPZ methodology in the LTR, when supported by design-specific and site-specific information and appropriately implemented by the COL applicant, is an acceptable approach for determining the plume exposure EPZ size; and
- identification of any issues related to the EPZ technical basis that are to be resolved prior to or as part of the COL proceeding.

To aid in the NRC's review, each section of the topical report individually identifies the approval request and associated acceptance criteria.

The EPZ methodology, as proposed in this LTR and to be implemented with detailed design information as part of a COL application, is a complete and sufficient approach for developing the basis for and specifying the size of the plume exposure EPZ for an LWR NuScale SMR design. The methodology is applicable to any plume exposure EPZ size, including the site boundary. The final EPZ size is the smallest distance at which the dose consequences of all screened-in accident sequences are less than their respective dose criteria. Based on the results of applying the methodology, the final plume exposure EPZ size may be different from the current 10 mile requirement.

The following summarizes the NuScale methodology for the technical basis for plume exposure EPZ size:

1. Dose criteria for the NuScale EPZ methodology have been defined based on the original EPZ basis and on EPA guidelines. These dose criteria are summarized in Table 5-1.

Table 5-1 Summary of dose criteria for NuScale EPZ methodology

Accident Type	Dose Criteria
DBA	1 and 5 rem TEDE
Less Severe	1 and 5 rem TEDE
More Severe	200 rem whole body acute

2. A risk-informed methodology for determining less severe and more severe accident sequences to be evaluated has been defined, along with the methodology for source term and dose evaluation.
3. The accident selection methodology uses PRA information and a plant-level, qualitative evaluation confirmation of defense-in-depth.
4. The methodology includes all internal and external events as well as all operating power levels including low power and shutdown.
5. A methodology for addressing multi-module risk has been developed which focuses on multi-module risks associated with common initiating events and structures, as well as shared systems between modules. The multi-module methodology is applicable to designs which have multiple modules.
6. Acceptance criteria to disposition other risks such as SFP accidents, severe accident phenomena, and security events have been provided in the methodology to determine whether these topics require further consideration in the EPZ technical basis. Example conclusions are provided based on the NuScale design, and a similar justification must be provided when applying the EPZ methodology.
7. A methodology for source term and dose evaluations has been defined that includes the appropriate application of software (such as RELAP5, MELCOR, and MACCS) for the events identified. This evaluation includes additional steps to evaluate doses within 500 meters and an integrated uncertainty analysis methodology.

6.0 References

6.1 Abstract and Section 1.0 Document References

6.1.1 *U.S. Code of Federal Regulations*, "Emergency Plans," Section 50.47, Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50.47).

6.1.2 *U.S. Code of Federal Regulations*, "Emergency Planning and Preparedness for Production and Utilization Facilities," Appendix E, Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50 Appendix E).

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There are no references in Section 5.0.

Enclosure 3:

Affidavit of Mark W. Shaver, AF-119991

NuScale Power, LLC

AFFIDAVIT of Mark W. Shaver

I, Mark W. Shaver, state as follows:

(1) I am the Director of Regulatory Affairs of NuScale Power, LLC (NuScale), and as such, I have been specifically delegated the function of reviewing the information described in this Affidavit that NuScale seeks to have withheld from public disclosure, and am authorized to apply for its withholding on behalf of NuScale

(2) I am knowledgeable of the criteria and procedures used by NuScale in designating information as a trade secret, privileged, or as confidential commercial or financial information. This request to withhold information from public disclosure is driven by one or more of the following:

(a) The information requested to be withheld reveals distinguishing aspects of a process (or component, structure, tool, method, etc.) whose use by NuScale competitors, without a license from NuScale, would constitute a competitive economic disadvantage to NuScale.

(b) The information requested to be withheld consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), and the application of the data secures a competitive economic advantage, as described more fully in paragraph 3 of this Affidavit.

(c) Use by a competitor of the information requested to be withheld would reduce the competitor's expenditure of resources, or improve its competitive position, in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

(d) The information requested to be withheld reveals cost or price information, production capabilities, budget levels, or commercial strategies of NuScale.

(e) The information requested to be withheld consists of patentable ideas.

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the methodology by which NuScale develops its technical basis for establishing NuScale's plume exposure emergency planning zone sizing.

NuScale has performed significant research and evaluation to develop a basis for this methodology and has invested significant resources, including the expenditure of a considerable sum of money.

The precise financial value of the information is difficult to quantify, but it is a key element of the design basis for a NuScale plant and, therefore, has substantial value to NuScale.

If the information were disclosed to the public, NuScale's competitors would have access to the information without purchasing the right to use it or having been required to undertake a similar expenditure of resources. Such disclosure would constitute a misappropriation of NuScale's intellectual property, and would deprive NuScale of the opportunity to exercise its competitive advantage to seek an adequate return on its investment.

(4) The information sought to be withheld is in the enclosed report entitled Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones. The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

(6) Pursuant to the provisions set forth in 10 CFR § 2.390(b)(4), the following is provided for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld:

(a) The information sought to be withheld is owned and has been held in confidence by NuScale.

(b) The information is of a sort customarily held in confidence by NuScale and, to the best of my knowledge and belief, consistently has been held in confidence by NuScale. The procedure for approval of external release of such information typically requires review by the staff manager, project manager, chief technology officer or other equivalent authority, or the manager of the cognizant marketing function (or his delegate), for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside NuScale are limited to regulatory bodies, customers and potential customers and their agents, suppliers, licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or contractual agreements to maintain confidentiality.

(c) The information is being transmitted to and received by the NRC in confidence.

(d) No public disclosure of the information has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to NRC, have been made, or must be made, pursuant to regulatory provisions or contractual agreements that provide for maintenance of the information in confidence.

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would be difficult for others to duplicate the technology without access to the information sought to be withheld.

I declare under penalty of perjury that the foregoing is true and correct. Executed on 6/10/2022.

Mark W. Shaver

Mark W. Shaver