March 12, 2018

Docket No. PROJ0769

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 "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," Revision 1, TR-0915-17772

REFERENCES: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Topical Report TR-0915-17772, 'Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites,' Revision 0" dated December 22, 2015 (ML15356A842)

NuScale Power, LLC (NuScale) hereby submits Revision1 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 methodology presented in TR-0915-17772 revision 1.

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". 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 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810. Enclosure 2 contains the nonproprietary version of the report entitled "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites".

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

Please contact Steven Mirsky at 240-833-3001 or at smirsky@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
Vice President, Regulatory Affairs
NuScale Power, LLC

Distribution: Frank Akstulewicz, NRC, OWFN-8-H4A
Samuel Lee, NRC, OWFN- 8G9A
Prosanta Chodhury, NRC, OWFN- 8G9A

Enclosure 1: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites", TR-0915-17772-P, Revision 1, proprietary version

Enclosure 2: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites", TR-0915-17772-NP, Revision 1, nonproprietary version

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0218-58900

Enclosure 1:

"Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," TR-0915-17772, Revision 0, Proprietary version

Enclosure 2:

"Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," TR-0915-17772, Revision 0, Non Proprietary version

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

March 2018

Revision 1

Docket: PROJ0769

NuScale Nonproprietary

NuScale Power, LLC

1100 NE Circle Blvd., Suite 200

Corvallis, Oregon 97330

www.nuscalepower.com

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

Department of Energy Acknowledgement and Disclaimer

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

CONTENTS

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

TABLES

FIGURES

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

TR-0915-17772-NP
Rev. 1

Licensing Topical Report

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

TR-0915-17772-NP
Rev. 1

Abstract

The purpose of this licensing topical report (LTR) is to provide the technical basis for the plume exposure pathway emergency planning zone (EPZ) sizing methodology for the NuScale small modular reactor (SMR) plant design. 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 design-specific 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 in the NuScale design, and to consider a consequence orientation in the approach. The screening of accident sequences includes the use of quantitative insights from the NuScale design-specific 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 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 that is appropriate for the NuScale design. A nuclear power plant using NuScale's SMR design comprises individual NuScale Power Modules producing approximately 5 percent of the thermal power of existing large plants, each with its own combined integral containment vessel (CNV) and reactor pressure vessel (RPV), and a dedicated turbine-generator set. The NuScale plant is of a scalable design where as many as 12 modules can be sequentially added. The LTR methodology was developed for a 12-module NuScale plant, but it is also applicable to a site with less than 12 modules.

The main body of the LTR contains the design-specific plume exposure EPZ size methodology for which NRC approval is being sought. To aid in the NRC's review and to illustrate how the EPZ size methodology would be used by future applicants, example source term and dose evaluations and example assessments of appropriate accident sequences to be evaluated are included in Appendices A, B, and C. NuScale is not seeking NRC approval of the information in the appendices, as the request for approval of EPZ size will be part of the combined license (COL) application.

The topical report requests an NRC review of the NuScale design-specific 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 (SER) 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 COL applicant, is an acceptable approach for justifying the plume exposure EPZ size for the NuScale design.

2. Identification of any issues related to the NuScale EPZ technical basis that are to be resolved prior to or as part of the COL review process.

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

TR-0915-17772-NP
Rev. 1

Executive Summary

The purpose of this LTR is to provide a methodology to establish the technical basis for plume exposure EPZ sizing for the NuScale design. 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 of 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. The NRC is also currently pursuing a rulemaking activity on emergency planning for SMRs (Rulemaking for Emergency Preparedness for Small Modular Reactors and Other New Technologies: Regulatory Basis, Reference 6.1.10), which includes EPZ considerations. As the NuScale SMR is in alignment with the new NRC rulemaking activity, NuScale 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 an SER on the design-specific plume exposure EPZ sizing methodology. The methodology herein, when supported by design-specific information and appropriately implemented by the COL applicant, is an acceptable approach to plume exposure EPZ sizing.

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 established utilizing design-specific PRA information supported by a comprehensive evaluation of severe accident sequences. It is applicable to: internal events, external hazards, all operating power levels, and all modes of operation.

The main body of the LTR, in Section 3.0, presents the design-specific EPZ size methodology for which NRC approval is sought. The methodology includes compilation of all accident sequences from the PRA and screening of the sequences for inclusion in the EPZ technical basis based upon multiple criteria. To aid in the NRC's review and to illustrate how the EPZ size methodology would be used by future applicants, example source term and dose evaluations and example assessments of appropriate accident sequences to be evaluated are included in Appendices A, B, and C. NuScale is not seeking NRC approval of the information in the appendices, as the request for approval of EPZ size will be part of the COL application.

The methodology first determines the appropriate sequences to be evaluated for EPZ in the NuScale design. The screening of accident sequences includes the use of quantitative insights from the NuScale design-specific PRA 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

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

TR-0915-17772-NP
Rev. 1

versus distance, serve as the basis for a plume exposure EPZ size methodology corresponding to the NuScale design.

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 design-specific, 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. The example results in the appendices, although not subject to approval, indicates that NuScale accident sequences are very infrequent and, even if such accidents occur, would not be expected to produce significant off-site consequences.

Finally, multi-module accidents are addressed in the NuScale EPZ methodology. The multi-module accident methodology focuses on multi-module risks associated with shared initiating events and structures, as well as shared systems among modules, which are unique to the NuScale design.

In summary, the NuScale methodology for establishing the design-specific technical basis for plume exposure EPZ sizing considers source terms and dose consequences. The methodology, when implemented with design information as part of a COL application, will be complete and sufficient to develop a basis for and to specify the size of the plume exposure EPZ for a NuScale plant. The methodology is applicable to any EPZ size, including the site boundary. The final EPZ size is the largest distance at which the dose consequence of each screened-in accident sequence is less than its respective dose criterion. Based on the results of applying the methodology, the final EPZ size may be different from the current 10 mile requirement.

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

TR-0915-17772-NP
Rev. 1

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 by combined license (COL) applicants using the NuScale small modular reactor (SMR) to establish the design-specific and site-specific plume exposure emergency planning zone (EPZ) size. 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 (SER) on the design-specific plume exposure EPZ sizing methodology.

1.2 Scope

This report provides a design-specific methodology for determining an appropriate plume exposure EPZ for a NuScale plant. 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], 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 appropriate accident sequences to be evaluated

- multi-module events and external events

- analysis of uncertainties

The main body of the LTR contains the design-specific plume exposure EPZ size methodology for which NRC approval is sought. To illustrate how the EPZ size methodology would be used by future applicants, example source term and dose evaluations, as well as example assessments of appropriate accident sequences to be evaluated, are included in Appendices A, B, and C. The information in the appendices is provided to facilitate: (1) NRC's review of the design-specific EPZ size methodology in the main body for which approval is sought; and (2) an understanding of how this LTR would be implemented by future applicants. NuScale is not seeking NRC approval of the information in the appendices, as it is purely for example, subject to change as the design matures, and the request for approval of EPZ size will be part of the COL application. This LTR is not part of the NuScale design certification application (DCA).

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

TR-0915-17772-NP
Rev. 1

1.3 Abbreviations and Definitions

Table 1-1.   Abbreviations

Term	Definition
AOP	abnormal operating procedure
ARP	alarm response procedure
ATD	atmospheric transport and dispersion
ATWS	anticipated transient without scram
BDBE	beyond-design-basis event
BDG	backup diesel generator
CCDP	conditional core damage probability
CCF	common-cause failure
CCFP	conditional containment failure probability
CDF	core damage frequency
CFDS	containment flooding and drain system
CNV	containment vessel
COL	combined license
CVCS	chemical and volume control system
DBA	design-basis accident
DBST	design-basis source term
DCA	design certification application
DCF	dose conversion factor
DHRS	decay heat removal system
EAL	emergency action level
ECCS	emergency core cooling system
EDMG	extensive damage mitigating guideline
ELAP	extended loss of AC power
EOP	emergency operating procedure
EPA	Environmental Protection Agency
EPZ	emergency planning zone
FEMA	Federal Emergency Management Agency
FSAR	final safety analysis report
HCLPF	high confidence of low probability of failure
HFE	human factors engineering
HSI	human-system interface
IAB	inadvertent actuation block
INSAG	International Nuclear Safety Advisory Group
ISG	interim staff guidance
LERF	large early release frequency
LOCA	loss of coolant accident
LOLA	loss of large areas

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

TR-0915-17772-NP
Rev. 1

Term	Definition
LOOP	loss of off-site power
LRF	large release frequency
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
PCT	peak cladding temperature
PGA	peak ground acceleration
PRA	probabilistic risk assessment
RBC	reactor building crane
RCS	reactor coolant system
RG	regulatory guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RVV	reactor vent valve
RXB	reactor building
SAMG	severe accident management guideline
SER	safety evaluation report
SFP	spent fuel pool
SGTF	steam generator tube failure
SMA	seismic margins assessment
SMR	small modular reactor
SOARCA	state-of-the-art reactor consequence analyses
SRM	staff requirements memorandum
SRO	senior reactor operator
SSC	structure, system, and component
SSE	safe-shutdown earthquake
TAF	top of active fuel
TEDE	total effective dose equivalent
UHS	ultimate heat sink

Table 1-2.   Definitions

Term	Definition
abnormal operating procedures	Procedures that are implemented under off-normal operational states which, because of appropriate design provisions, would most likely not

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

TR-0915-17772-NP
Rev. 1

Term	Definition
	result in the loss of a critical safety function, cause any significant damage, nor lead to accident conditions.
anticipated operational occurrences	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.
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.
design-specific PRA	For a nuclear power plant at the design certification or combined operating license stage, where the plant is not built or operated, the design-specific PRA model reflects the as-designed plant.
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.
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. Events such as earthquakes, tornadoes, and floods from sources outside

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

TR-0915-17772-NP
Rev. 1

Term	Definition
	the plant and fires from sources inside or outside the plant are considered external events.
FLEX	An approach for adding diverse and flexible mitigation strategies for mitigating and coping with beyond-design-basis events.
high confidence of low probability of failure	A measure of seismic capacity of a structure, system, or component, expressed in terms of a threshold earthquake intensity, below which failure of the structure, system, or component is highly unlikely.
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.
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.
safe-shutdown earthquake	The maximum earthquake for which certain structures, systems, and components are designed to remain functional.
Seismic Category I	Structures, systems, and components that are designed to remain functional if a safe-shutdown earthquake (SSE) 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.
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.
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 head sink pools are located below grade in the reactor building.

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

TR-0915-17772-NP
Rev. 1

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 for the NuScale design, and discuss recent industry and NRC documents that address reevaluation of EPZ size and planning elements for SMRs, including an upcoming 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 6.2.1) 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 accidents could be expected to exceed the applicable protective action guides (PAG) of 1 and 5 rem.

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

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

TR-0915-17772-NP
Rev. 1

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 (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) (References 6.2.6 and 6.2.7, respectively). The staff reviewed these exemption requests against the requirements in 10 CFR 50.47, (Reference 6.1.1); 10 CFR 50, Appendix E (References 6.1.2); and Emergency Plans, 10 CFR 72.32 (Reference 6.2.8).

Industry believes that siting and building SMRs with appropriate EPZ size and planning elements will have benefits for all stakeholders. This is based on the expectation that the SMR 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 very 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, a NuScale Power Module (NPM) is approximately 5 percent of the thermal power of existing large plants, which translates into a much lower fission product inventory. In addition, the NuScale 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.

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

The NuScale design offers 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.

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

TR-0915-17772-NP
Rev. 1

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). NRC approval of an early site permit 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 (References 6.2.10, 6.1.6, and 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.

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

TR-0915-17772-NP
Rev. 1

- 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 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

Two recent NRC documents, SECY-15-0077 and SECY-16-0069, address EP-related rulemaking for SMRs.

In SECY-15-0077 (Reference 6.2.3), the staff seeks 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

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

TR-0915-17772-NP
Rev. 1

ensure that applicable requirements for adequate protection of public health and safety, and the environment, are met. In a staff requirements memorandum (SRM) dated August 4, 2015, 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 and ending in April 2020 when the final rule is published. In the SRM dated June 22, 2016 (Reference 6.2.16), the Commission approved the staff's proposed schedule.

The most recent NRC SMR-related document is the 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.

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, 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 less severe and more severe accident sequences, and a risk-informed assessment of the credibility of these accident sequences, in order "to determine appropriate accidents to be evaluated" (Reference 6.2.3) for the EPZ basis.

- the risk-informed assessment of accident sequences applies use of accident sequence frequency information from the NuScale design-specific probabilistic risk assessment (PRA), confirmed by a sequence-specific assessment of defense-in-depth.

- 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.

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

TR-0915-17772-NP
Rev. 1

- 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 events and hazards and all operating power levels, including low power and shutdown, multi-module accidents, and spent fuel pool (SFP) accidents to provide assurance of completeness.

- a design-specific methodology to assess uncertainties as confirmation of analytical results.

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

TR-0915-17772-NP
Rev. 1

3.0   Design-Specific Screening Methodology for Accidents

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 size. Sections 3.3, 3.4, and 3.5 discuss the methodology for using a risk-informed approach to select appropriate accident sequences to include in the EPZ technical basis. Sections 3.6 to 3.9 discuss the methodology for addressing seismic risk, multi-module risk, other PRA risks, and security events, respectively. 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 in list format as delineated below with corresponding LTR section number:

- Compile accident sequences from the PRA for all initiators (Section 3.4.1)

- Perform seismic high confidence of low probability of failure (HCLPF) screening (Section 3.6)

- Calculate total core damage frequency (CDF) (Section 3.4.1)

- Perform accident sequence screening based on frequency (Section 3.4.2)

- Perform additional assessment of defense-in-depth as necessary to substantiate the low accident sequence frequency and confirm the associated sequence screening (Section 3.5)

- Characterize screened-in sequences as more or less severe based on whether containment is intact or bypassed (Section 3.4.3)

  - If no less severe accident sequence was screened-in, include the highest likelihood screened-out less severe sequence in the EPZ technical basis (Section 3.4.3)

- Perform multi-module screening and add any screened-in multi-module accidents (Section 3.7)

- Perform severe accident simulations of screened in accident sequences to determine environmental source term (Sections 4.1.1 and 4.2)

- Add the design-basis source term (DBST) (Section 3.3)

- 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 and adjust MACCS results inside 0.5 km as necessary (Section 4.2.4)

- Perform uncertainty analysis and justify important parameters (Section 4.3)

  - Repeat severe accident and consequence simulations if necessary

- Determine the final EPZ distance as the largest distance among the following, with the site boundary as a minimum:

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

TR-0915-17772-NP
Rev. 1

  – The larger 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 larger 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 the conditional probability of exceeding 200 rem whole body acute dose drops below 1E-3 for screened-in more severe accident sequences (Section 4.2.3)

Overall EPZ
Process
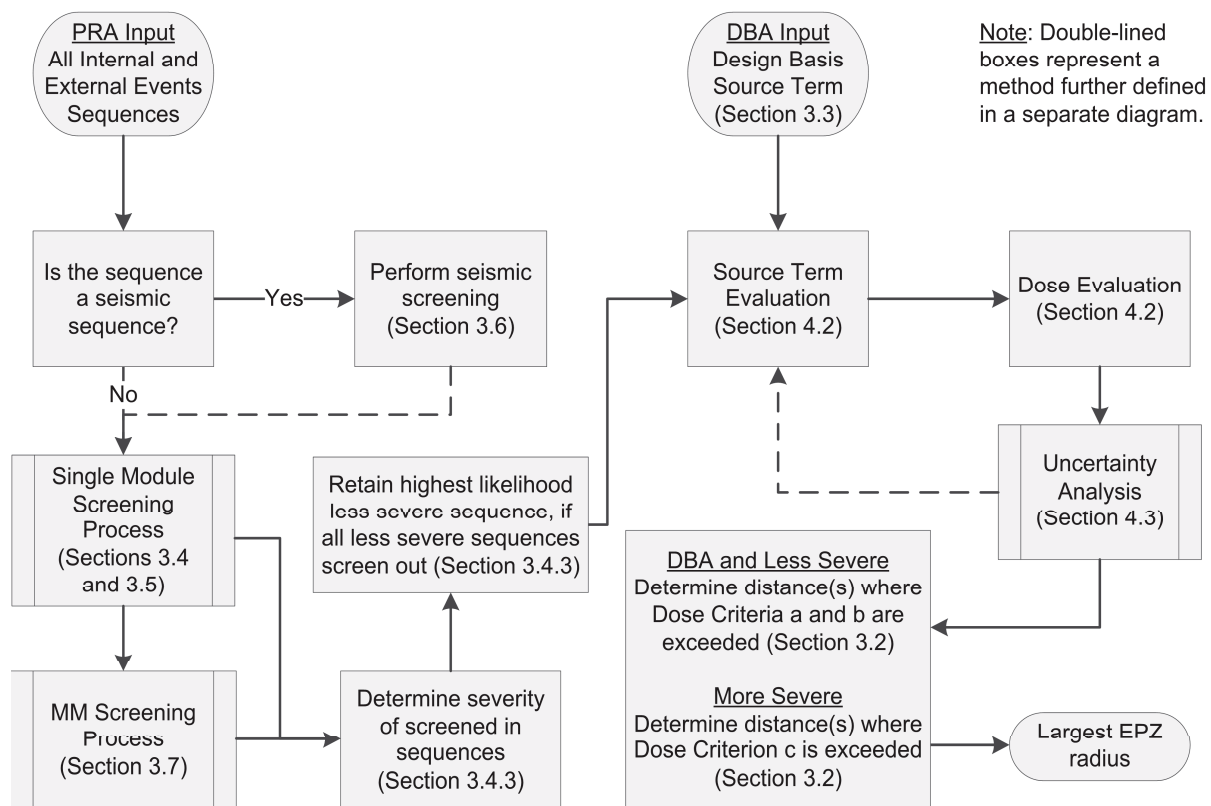
Figure 3-1. Overall methodology to determine EPZ distance

An example implementation of portions of the Sections 3.4 and 3.5 methodologies for determining appropriate accident sequences to be evaluated for EPZ is presented in Appendix C. This is for illustration only and, as previously stated, NuScale is not seeking NRC approval of the information in the appendices.

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

TR-0915-17772-NP
Rev. 1

3.1   Assumptions

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

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

- PRA development, including plant-specific PRAs performed by licensees and NRC over the last 20 years, the recent SOARCA study (Reference 6.2.14), and evolution of PRA industry-consensus standards, Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications (Reference 6.3.1) 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 (An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, Regulatory Guide [RG] 1.174 [Reference 6.3.2]), the reactor oversight process, and new reactor licensing where all new designs are required to perform a design-specific 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.

- 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 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 design-specific 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.

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

TR-0915-17772-NP
Rev. 1

- SECY-15-0077 (Reference 6.1.7) 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.

Assumption 2: In the NuScale risk-informed approach to EPZ, balance should be maintained between defense-in-depth and risk considerations.

Justification: Risk-informed processes for any regulatory application, in particular EPZ sizing, should combine and balance insights from a deterministic assessment of the adequacy of defense-in-depth with quantitative risk insights from the PRA. 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).

Defense-in-depth considerations are being applied in the NuScale risk-informed approach in a sequence-specific manner as part of a risk-informed approach to determine the appropriate accident sequences to be evaluated.[1]

---

[1] In SECY-15-0077 recommendation to revise regulations and guidance related to emergency planning for SMRs through rulemaking, the staff stated examples of some broad issues that are likely to arise while developing this emergency planning framework, some of which may require future Commission direction. One of these issues was "determining appropriate accidents to be evaluated."

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

TR-0915-17772-NP
Rev. 1

Risk-Informed Framework

Traditional "Deterministic" Approach

- Unquantified probabilities
- Design-basis accidents
- Defense in depth and safety margins
- Can impose unnecessary regulatory burden
- Incomplete

Risk-Informed Approach

- Combination of traditional and risk-based approaches through a deliberative process

Risk-Based Approach

- Quantified probabilities
- Thousands of accident sequences
- Realistic
- Incomplete

Figure 3-2. Risk-informed framework

Addressing EPZ using a risk-informed approach for NuScale offers an opportunity to optimize EPZ size and the basis for this size 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 for existing plants were not based on quantification of accidents, but rather "were qualitatively found adequate as a matter of judgment" (Reference 6.1.8). 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 a balance between quantitative and qualitative methods. In the nearly four decades 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 balanced approach for justification of NuScale EPZ size.

Assumption 3: A dose-based approach with a consequence orientation is appropriate for use in the NuScale EPZ size basis.

Justification: NuScale intends to implement the risk-informed approach in a way that addresses NRC guidance and applicable, historical concepts for EPZ development. This guidance and applicable historical concepts include:

- applying a dose-based framework with a consequence orientation (SECY-15-0077)

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

TR-0915-17772-NP
Rev. 1

- 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)

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. Uncertainties associated with state of knowledge limitations and with hazards and events not easily amenable to treatment in PRA have often led to overly conservative solutions and unrealistic accidents. NuScale's risk-informed approach includes steps to evaluate and limit uncertainties 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

- use of state-of-the-art methods to calculate source terms and doses, which greatly reduce uncertainty as compared to previous quantitative methods, which were excessively conservative

  – integrated uncertainty analysis is also used to increase confidence in the best-estimate 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 face of very low frequency events including:

  – use of a multiplier on total CDF and extension to very low frequencies so as to assure a wide range of frequency screening, beyond what has traditionally been considered credible in severe accident evaluations (Section 3.4.2)

  – application of methods for deterministic assessment of accident sequence defense-in-depth so as to substantiate the low sequence frequencies and confirm the accident sequence screening

  – provision for deterministically-based, operationally-focused mitigation capability as addressed in Appendix E

  – site emergency plans for a NuScale facility 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, which is deterministically-based

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

3.2 Dose-Based Criteria

The dose-based criteria for EPZ, based on the NUREG-0396 sizing rationale and used as input to the determination of the generic distance for the plume exposure EPZ for existing plants, are restated below. Per the NEI white paper, industry considers these dose-based criteria to be appropriate for SMR EPZ plume exposure sizing.

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

TR-0915-17772-NP
Rev. 1

- Criterion a: The EPZ should encompass those areas in which projected dose from design-basis accidents (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 to 5 rem[2] TEDE (Reference 6.1.4). Throughout the remainder of the report, whenever "PAGs" is used, it refers only to the early phase PAGs. 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, 3.4, and 3.5, and method details for applying the dose criteria are provided in Section 4.2.

NuScale has addressed the cumulative plant risk design objectives as specified in the NEI white paper. Addressing cumulative plant risk design objectives is necessary since the evaluation against dose-based criteria discussed above addresses only individual sequences. Although these are not dose-based, the criteria have been linked to cancer and fatality risks and are appropriate to consider. It is also necessary to ensure that the total plant risk does not exceed appropriate objectives.

These objectives are as follows:

- total mean core damage frequency (CDF) < 1E-5 per plant year

- total mean large release frequency (LRF) < 1E-6 per plant year

The design-specific PRA has been used to demonstrate that the plant risk design objectives are met for internal and external events as well as all plant operating states for the NuScale design. The COL applicant should confirm that the plant risk design objectives are met with the plant-specific PRA.

3.3   Determination of Appropriate Design-Basis Accidents to Be Evaluated

For Criterion a (Section 3.2), the methodology to be applied in the determination of the accident to be evaluated will be based on the DBST from Chapter 15 of the applicant's final safety analysis report (FSAR). The DBST is a surrogate release from the

---

[2] 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.

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

TR-0915-17772-NP
Rev. 1

containment used to assess off-site dose. {{

}}[2(a),(c)] The DBST will be evaluated for the Criterion a comparison against the PAGs as discussed in Section 4.2. The DBST is specifically representative of a single module accident and is not appropriate for multi-module considerations. The EPZ distance as calculated by the DBST source term will be compared against the EPZ distance from 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 RG 1.183, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Reference 6.3.7) source term. The DBST in Chapter 15 includes a release into containment, but additional analysis will be performed as part of Chapter 15 to evaluate fission product transport in containment and fission product leakage from containment to the environment. The COL applicant should use the results of this additional analysis so as to make the DBST consistent with the EPZ source term definition.

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

For Criteria b and c (Section 3.2), the methodology of this section is used to screen accident sequences for inclusion in the EPZ technical basis. Once appropriate sequences are determined, the evaluation of these sequences involve 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. This methodology includes accident sequences from all internal events, external events, and operating modes. It utilizes a risk-informed process, including both frequency and defense-in-depth considerations, as justified by Assumptions 1 and 2 in Section 3.1.

Figure 3-3 provides a high-level overview of the screening methodology. There are three main elements of the method: (1) initial sequence compilation, (2) accident sequence screening based on frequency and defense-in-depth, and (3) final classification of severity. Each of these elements is described in detail in Sections 3.4.1, 3.4.2, and 3.4.3, respectively.

The EPZ methodology initially requires the following hazard models from the site-specific PRA: all internal and external events and hazards and all operating power levels including low power and shutdown. A seismic PRA may be included, pending the result of initial seismic screening using the seismic margin assessment (SMA). If any other external event not listed is included in the site-specific PRA, it would also be required. A separate multi-module process for EPZ is detailed in Section 3.7.

Due to NuScale's low risk profile, this method utilizing multiple frequency screening bins is appropriate as compared to a single frequency-based cut-off, such as that used in the SOARCA project (Reference 6.2.14). There is limited regulatory guidance

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

TR-0915-17772-NP
Rev. 1

and precedence on frequency screening that is appropriate for an advanced reactor with such a low risk profile. This method with a wide frequency screening, which includes threshold based on a multiplier on the total CDF, is intended to produce an acceptable spectrum of accident sequences that would have consequences evaluated in order to determine EPZ size. This ensures a consequence orientation to the methodology, consistent with Assumption 3 in Section 3.1.

{{

}}[2(a),(c)]

Figure 3-3. Single module risk-informed screening process to determine credible accident sequences

3.4.1 Compilation of Accident Sequences

All internal events, external events, and operating modes are considered together. {{

}}[2(a),(c)]

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

TR-0915-17772-NP
Rev. 1

{{

}}[2(a),(c)]

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 EPZ methodology. This is necessary to retain enough independence for assessment of defense-in-depth. Appendix C provides an example of NuScale's PRA sequences.

{{

}}[2(a),(c)] There is a separate screening criterion for seismic sequences, discussed in Section 3.6. Based on the results of this screening, seismic sequences will either screened out of consideration, or included for screening based on frequency the same as other sequences.

{{

}}[2(a),(c)]

The total CDF and individual sequences frequencies are used as the inputs to the screening criteria. In this case, the total CDF would be the sum from the entire PRA.

3.4.2 Screening of Single Module Accidents

There are three possibilities for screening accident sequences based on frequency:

1. {{

}}[2(a),(c)]

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

TR-0915-17772-NP
Rev. 1

3. {{

}}²⁽ᵃ⁾,⁽ᶜ⁾

The CDF values used for screening should have the same number of significant figures as are used in the PRA. It is noted that 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) and SOARCA (Reference 6.2.14). The Figure 3-3 frequency screening bins have been chosen based on per module year. 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 3.7.

{{

}}²⁽ᵃ⁾,⁽ᶜ⁾

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

TR-0915-17772-NP
Rev. 1

{{

}}[2(a),(c)] For perspective, the probability associated with one in a billion years is orders of magnitude lower than the probability of civilization ending, catastrophic events such as meteor strikes. Appendix C contains an example of calculating the total CDF.

3.4.3   Final Classification of Accidents by Severity

The final step of the methodology is to classify all screened-in accident sequences by containment integrity. If the sequence does not include a loss of containment integrity (i.e., core damage is not associated with containment failure), the accident sequence is classified as "less severe" and assessed against the dose criteria of Section 3.2. The effect of an intact containment is that the only potential radionuclide release to the environment is by nominal containment leakage. The maximum allowable containment leakage is 0.2 percent volume/day, which is why intact containment can be equated with "less severe." If the methodology results in the screening out of every

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

TR-0915-17772-NP
Rev. 1

less severe accident sequence, the highest frequency intact containment accident sequence will be retained for source term and dose evaluation.

If containment integrity is not maintained (i.e., containment bypass loss-of-coolant accident (LOCA), containment isolation fails, or containment is otherwise breached) the accident sequence is classified as "more severe" and dose evaluation is performed using the methodology of Section 4.2.3.

The NuScale containment has been designed to reduce the possibility of containment failure, consistent with regulatory expectations for advanced reactors. For example, there are multiple isolation valves on all containment piping penetrations. Additionally, the main steam and feedwater piping is rated for RPV design pressure up to the second isolation valve. Due to this design, it is appropriate to consider any accident sequence with a loss of containment integrity to be more severe.

Generally, the "more severe" sequences would also include those cases where the containment has failed as a result of severe accident phenomena. However, in the NuScale design certification PRA, no physically credible containment failure mechanism (other than bypass) from severe accident phenomena could be identified. Hence in this method, "bypass" is sometimes used to denote failure of the containment function since it is the primary mechanism by which the containment function fails.

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 (e.g., a dropped module may or may not result in a breached containment), the accident sequence should be considered both less and more severe. In this case, analyses against both dose criteria shall be performed according to the following methods.

- {{

}}[2(a),(c)]

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

TR-0915-17772-NP
Rev. 1

3.5 Methodology for Sequence-Based Assessment of Defense-in-Depth

3.5.1 Introduction

This section addresses the methodology for an assessment of defense-in-depth for core damage accident sequences which, based on plant-specific PRA information and the Section 3.4 determination of appropriate accident sequences to be evaluated for EPZ, are in the frequency range from {{
}}[2(a),(c)] As described in Section 3.4, {{

}}[2(a),(c)]

In identifying accident sequences to be evaluated as part of the basis for NuScale EPZ size, the methodology is applied to core damage sequences. In this context, a sequence refers to an event tree pathway that follows from a particular initiating event, through system and operator response, to the end state of core damage for a single module. Multi-module accidents are addressed in Section 3.7. Several of the defense-in-depth attributes such as initiating event frequency and core damage frequency are easily determined from the core damage sequence. It is important to note, however, that other parts of the PRA analysis are needed to evaluate other defense-in-depth attributes. As such, assembling information on the conditional containment failure probability (CCFP), time to core damage, secondary confinement structures, and other mitigation measures along with the core damage sequence results will greatly facilitate the defense-in-depth evaluation.

Assessing defense-in-depth for specific accident sequences is a process for which there is little regulatory guidance or precedent. Existing guidance is for a broad assessment as to whether the defense-in-depth philosophy is supported on a plant-wide basis. Thus, NuScale has adapted existing guidance to develop the methodology for a sequence defense-in-depth assessment to support the accident sequence screening process for EPZ sizing. The adaptation is necessary to ensure that each defense-in-depth criterion and attribute can be evaluated at the sequence level. The primary basis is International Nuclear Safety Advisory Group (INSAG), Defence in Depth in Nuclear Safety, INSAG-10 (Reference 6.3.13). However, Section 3.5.3 shows the consistency between NuScale's methodology and RG 1.174 (Reference 6.3.2).

For accident sequence screening purposes, it can be asserted that defense-in-depth is reflected in the very low PRA frequencies for the sequences under consideration {{
}}[2(a),(c)] and that further work to assess defense-in-depth for the

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

TR-0915-17772-NP
Rev. 1

sequences is not necessary. However, NuScale recognizes the strong precedent of the defense-in-depth concept in the safety community and in regulatory guidance as a means to address uncertainty and unforeseen failure mechanisms or phenomena, as well as to provide deterministic engineering insights to complement the results and insights from the PRA. Hence, an assessment of sequence defense-in-depth is incorporated in the NuScale EPZ sizing methodology to provide confirmation of low frequency PRA results.

3.5.2 NuScale Attribute Evaluation-Based Methodology

The NuScale defense-in-depth methodology utilizes five criteria, along with evaluation of qualitative and quantitative attributes associated with each criterion, for assessing sequence defense-in-depth. The five criteria draw on the five INSAG-10 (Reference 6.3.13) levels of defense-in-depth and adapt these levels to produce a framework for sequence-based assessment to support reaching a conclusion as to whether there is consistency with the defense-in-depth philosophy. Additionally, Section 3.5.3 contains a summary of how the five criteria and supporting attributes of NuScale's methodology are consistent with the four layers of defense and the seven evaluation factors from RG 1.174.

The five criteria to support an accident sequence-based assessment of defense-in-depth are successive compensatory measures to prevent the accident initiation or mitigate fission product release if the sequence progresses to core damage. The five criteria are as follows:

- Criterion 1. Prevention of abnormal operation and failures which would initiate the accident sequence

- Criterion 2. Control of abnormal operation and detection of the failures associated with accident initiation so as to prevent further deterioration of plant status

- Criterion 3. Control of the accident sequence within the design basis so as to prevent core damage

- Criterion 4. Control of beyond-design-basis conditions, including limiting core damage progression and providing containment of any resulting fission product release from the damaged core

- Criterion 5. Mitigation of the consequences of releases of radioactive materials which could result from the accident sequence if the containment function is impaired

The assessment of defense-in-depth for sequence screening employs a risk-informed, performance-based approach, which was developed by NuScale. NuScale's purpose in developing this approach was to establish a practical means for assessing defense-in-depth on a sequence basis and to avoid relying solely on probabilistic criteria for screening sequences in evaluating the EPZ size.

The NuScale approach utilizes attributes in the form of risk metrics, design features, and operational features. Some of the attributes, particularly the risk metrics, are quantitative, and others are qualitative, with both types of attributes to be evaluated by the COL

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

TR-0915-17772-NP
Rev. 1

applicant to assess the extent to which each of the five criteria are addressed for each sequence requiring defense-in-depth assessment. Thus, while the approach relies on some engineering judgment, it involves both quantitative and qualitative attributes and associated attribute valuation. This is much more structured and transparent than a purely qualitative approach to assessing defense-in-depth.

Each defense-in-depth criterion includes attributes and each attribute is evaluated individually; evaluation of an attribute by the COL applicant consists of assigning a sequence-specific level of defense-in-depth for the attribute. A template for the attribute evaluation is shown in Table 3-1.

Table 3-1.   Template for attribute evaluation for a given defense-in-depth criterion

Defense-in-Depth Criterion X Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Attribute 1			
Attribute 2			
Attribute 3			
Attribute 4			

There are three levels of defense-in-depth as shown in the table:

High Defense-in-Depth – represents the highest valuation level of sequence-specific defense-in-depth in terms of substantiating the low frequency associated with the sequence in the PRA; the risk metrics and plant design features associated with this level (e.g., passive, automatic responses; diverse components and systems) would have margin beyond what can be considered as acceptable for advanced LWR designs

Medium Defense-in-Depth – represents the middle or nominal valuation level of sequence-specific defense-in-depth in terms of substantiating the low frequency associated with the sequence in the PRA; the risk metrics and plant design features associated with this level would qualitatively differ from those for the highest level (e.g., fewer passive features, more reliance on operator action) and, while not a rigid standard, would be commensurate with the higher expectations for safety commonly associated with advanced LWR designs

Low Defense-in-Depth – represents the lowest valuation level of sequence-specific defense-in-depth in terms of substantiating the low frequency associated with the sequence in the PRA; the risk metrics and plant design features associated with this level would be qualitatively lower than the higher levels and

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

TR-0915-17772-NP
Rev. 1

although not considered insufficient still require compensation for an overall assessment of adequacy

As noted, the three levels are intended to qualitatively differentiate the extent to which defense-in-depth is addressed for a given attribute. Also, the levels should be viewed in the context of advanced LWRs. That is, the middle or nominal level represents defense-in-depth which is commensurate with the higher expectations for safety commonly associated with advanced LWR designs, and the highest level represents margin beyond what would nominally be expected for advanced LWRs. The lowest level would be the lowest that would be expected for advanced LWRs and, as discussed in Section 3.5.2.6, only one of the five defense-in-depth criteria can be evaluated as low and still have the sequence under consideration be screened out of the EPZ technical basis.

In the five subsections below, which address the specifics of the methodology for each of the defense-in-depth criteria, the attributes to be considered for each criterion are defined, along with default ranges (or default values or features in the case of the qualitative attributes) for evaluation of the three levels. The attributes that are applicable for a given criterion could in general be different from one sequence to another (e.g., a sequence that is initiated by an internal event would not include the external hazard initiating event frequency attribute in the evaluation). When applying the methodology to determine a final EPZ size, the technical basis that determines each attribute valuation should be documented.

Assessment of a given defense-in-depth criterion (i.e., evaluating the extent to which the criterion has been met), will be done as follows:

- Criterion meets High Defense-in-Depth – no attributes are valued as Low, with at least one of the attributes valued as High

- Criterion meets Medium Defense-in-Depth

    – Does not meet High Defense-in-Depth, and

    – no attributes valued as Low, or

    – one attribute valued as Low, with at least one attribute valued as High

- Criterion meets Low Defense-in-Depth

    – Does not meet High or Medium Defense-in-Depth

After each of the criteria is assessed, an overall assessment is made of the defense-in-depth for the sequence under consideration as discussed in Section 3.5.2.6. The evaluation of the attributes for each criterion by the COL applicant shall be based on information from the NuScale plant design and the plant-specific PRA.

It is possible that a given attribute might not meet even Low Defense-in-Depth. Although not expected for an advanced LWR such as NuScale, if such a case were to arise, the sequence would automatically be screened into the set of accident sequences to be evaluated for EPZ.

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

TR-0915-17772-NP
Rev. 1

3.5.2.1 Criterion 1. Prevention of Abnormal Operation and Failures Which Would Initiate the Accident Sequence

In Criterion 1, the risk metric attributes are initiating event frequencies. That is, the likelihood of abnormal operation or of a failure leading to automatic reactor trip and initiation of the accident sequence under consideration (e.g., loss of feedwater, LOCA), or a manual trip prompted by conditions other than those involved in a normal shutdown. Two such attributes are shown in Table 3-2: (1) internal event initiating event frequency, and (2) external hazard initiating event frequency.

The defense-in-depth values for internal initiating event frequencies range from events that are not expected to occur within the plant lifetime for the high level to those that are expected to occur annually. Because external hazards such as earthquakes and tornadoes can impact the ability of plant systems to respond to an upset condition, the defense-in-depth values for external frequencies are lower. As an example, if a sequence initiated by loss of support systems (e.g. loss of instrument air) required assessment for defense-in-depth, and if an initiating event frequency of 1.6E-2 per module year from the PRA was applied, the COL applicant would assign Medium for the internal event initiating event frequency attribute in Table 3-2. If, however, based on the plant design and the site, this frequency was 6E-3 per module year in the plant-specific PRA, High would be assigned. It is also noted that the only one of the two attributes would be assessed for each sequence, depending on the initiating event.

Table 3-2.   Criterion 1: Prevention of abnormal operation and failures which would initiate the accident sequence

Defense-in-Depth Criterion 1 Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Internal event initiating event frequency (per module year)	$\leq$ 1E-2	> 1E-2 and $\leq$ 1	> 1
External hazard initiating event frequency (per year)	$\leq$ 1E-4	> 1E-4 and $\leq$ 1E-2	> 1E-2

3.5.2.2 Criterion 2. Control of Abnormal Operation and Detection of the Failures Associated with Accident Initiation to Prevent Further Deterioration of Plant Status

In Criterion 2, the attributes are qualitative and consider inherent plant design and operational features as well as systems to control and detect abnormal operations and failures such that they minimize the need for actuation of safety systems. In response to a deviation from steady-state operation resulting from the accident sequence initiating event, the attributes consider plant response and the systems designed to bring the plant back to normal operating conditions. The defense-in-depth metrics for safety and nonsafety-system response to initiating events range from completely passive systems to those that require local, manual control by plant operators; defense-in-depth is enhanced through the use of passive and highly-reliable, power-independent fail-safe

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

TR-0915-17772-NP
Rev. 1

safety systems. There are no risk metric attributes for Criterion 2. Specific ranges for attribute evaluation are listed in Table 3-3.

Table 3-3.   Criterion 2: Control of abnormal operation and detection of the failures associated with accident initiation to prevent further deterioration of plant status

Defense-in-Depth Criterion 2 Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Safety system response to detect and control initiating event[1]	Passive or fail-safe system	Active system with automatic control	Active system with manual control
Nonsafety-system response to detect and control initiating event[1]	Automatic system	Manual action from control room	Manual, local action

[1]In cases where a sequence involves more than one safety or nonsafety-system, each system is evaluated separately, and the evaluation with the highest defense-in-depth valuation is used for judging the attribute.

3.5.2.3 Criterion 3. Control of the Accident Sequence Within the Design Basis to Prevent Core Damage

In Criterion 3, given the need for safety system actuation, the focus is the prevention of core damage, with attributes addressing engineered safety features and protection systems to prevent the evolution of the sequence toward severe accidents. There are two quantitative risk metric attributes. The risk metrics are the focused PRA CDF for the sequence, which credits only safety systems to respond to the initiating event and the sequence conditional core damage probability (CCDP), which credits both safety and nonsafety-systems and components but is conditional on occurrence of the off-normal initiating event.

The high valuation for the focused PRA is when the safety systems alone are well above the quantitative objective established to meet the Commission's safety goal (i.e., total CDF of less than 1E-4 per reactor year), while the low valuation is within an order of magnitude of the goal. Similarly for CCDP, the defense-in-depth values were based on the sequence CDF values (including credit for nonsafety-system response), also compared to the safety goal. In both cases, it is appropriate for the low valuation to be greater than the safety goal as the goal is for total plant CDF and the valuation is performed at the sequence level. Specific ranges for Criterion 3 attribute evaluation are listed in Table 3-4.

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

TR-0915-17772-NP
Rev. 1

Table 3-4.   Criterion 3: Control of the accident sequence within the design basis to prevent core damage

Defense-in-Depth Criterion 3 Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Sequence CDF (per module year) considering only safety-related systems (focused PRA)	$\leq$ 1E-5	>1E-5 and $\leq$ 1E-3	>1E-3
Sequence conditional core damage probability (CCDP)	$\leq$ 1E-5	> 1E-5 and $\leq$ 1E-3	> 1E-3

3.5.2.4 Criterion 4. Control of Beyond-Design-Basis Conditions, Including Limiting Core Damage Progression and Providing Containment of any Resulting Fission Product Release from the Damaged Core

In Criterion 4, given that core damage has occurred, the focus is on limiting core damage progression and on limiting fission product release from the containment, and, in the case of a containment bypass sequence, from the reactor coolant system (RCS) and interfacing system piping. As such, the attributes used to assess this criterion emphasize the containment function. There is one quantitative risk metric attribute, two quantitative timing attributes, and one qualitative attribute.

The risk metric is the sequence CCFP, given core damage. The high valuation for CCFP covers sequences in which there is considerable margin to meet the NRC's Standard Review Plan, NUREG-0800 Chapter 19.0 (Reference 6.3.18) acceptance criteria for containment failure (i.e., LRF/CDF ≤ 0.1); the low valuation covers sequences that do not meet the criteria.

The timing attributes are time to beginning of core damage, and the coping time for loss of all AC power. The threshold values for the time to core damage consider the possibility of additional resources becoming available to limit core damage progression; medium valuation provides time for emergency operating facility staffing (as described in Functional Criteria for Emergency Response Facilities, NUREG-0696 [Reference 6.3.19]) and low valuation limits the control of accident progression to the operator recovery actions that would be performed in accordance with the EOPs (as described in Section E.3).

For sequences involving a complete loss of all AC power only, the coping time is the time from the onset of the accident to the time when AC power is needed to be restored to maintain adequate core cooling. The coping time for high valuation is based on expectations for passive plants; it is also the time in which outside resources are expected to be available to support FLEX strategies. The coping time for medium valuation is based on expectations for restoring AC power; it is also the time in which site access is expected to be restored to support FLEX strategies.

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

TR-0915-17772-NP
Rev. 1

The one qualitative attribute is containment isolation response. The defense-in-depth metrics for containment isolation range from a high valuation of highly-reliable fail-safe components (e.g., valves that fail closed on a loss of power) to a low valuation of only relying on check valves that historically have not been as reliable for leak-tightness.

Criterion 4 is mainly a control criterion as opposed to Criterion 5 which is a mitigation criterion. Specific ranges for Criterion 4 attribute evaluation are listed in Table 3-5.

Table 3-5.   Criterion 4: Control of beyond-design-basis conditions, including limiting core damage progression and providing containment of any resulting fission product release from the damaged core

Defense-in-Depth Criterion 4 Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Sequence conditional containment failure probability (CCFP)	≤ 0.01	> 0.01 and ≤ 0.1	> 0.1
Time to beginning of core damage (hours)	≥ 8	< 8 and ≥ 1	< 1
Coping time (only for loss of all AC power sequences) (hours)	≥ 72	< 72 and ≥ 24	< 24
Containment isolation response	Fail-safe actuated valves	Active actuated valves	Only check valves

3.5.2.5 Criterion 5. Mitigation of Consequences of Releases of Radioactive Materials Which Could Result from the Accident Sequence if the Containment Function is Impaired

Criterion 5 includes attributes aimed at controlling the course of a severe accident and mitigating the consequences given that the containment function is impaired. Attributes include plant features and accident management measures that would prevent or reduce radioactive release to the environment. The criterion includes two attributes: (1) a risk metric for LRF for the sequence, and (2) an attribute for secondary confinement.

The LRF value for high defense-in-depth covers sequences in which there is considerable margin to meet the quantitative objective established to meet the Commission's safety goal (i.e., total LRF of less than 1E-6 per reactor year); the value for low defense-in-depth covers sequences that fall short of meeting the safety goal. It is conservative to apply a safety goal for total LRF for a sequence level evaluation. The metrics for secondary confinement range from a high valuation of a Seismic Category 1 structure which would provide some mitigation to a low valuation of no secondary confinement. Specific ranges for Criterion 5 attribute evaluation are listed in Table 3-6.

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

TR-0915-17772-NP
Rev. 1

Table 3-6.   Criterion 5: Mitigation of the consequences of releases of radioactive materials which could result from the accident sequence if the containment function is impaired

Defense-in-Depth Criterion 5 Attributes	Attribute Valuation Ranges		
	High Defense-in-Depth	Medium Defense-in-Depth	Low Defense-in-Depth
Sequence LRF (per module year)	$\leq$ 1E-8	> 1E-8 and $\leq$ 1E-6	> 1E-6
Secondary confinement[1]	Seismic Category I	Seismic Category II or Nonseismic	None
1. Secondary confinement should only be considered as an additional attribute if it has not already been included in the calculation of LRF. The intent here is to capture "additional" design features that would not otherwise be considered and may support operationally-focused mitigation actions.			

3.5.2.6 Overall Attribute Evaluation-Based Defense-in-Depth Assessment

After each of the criteria is assessed, an overall assessment is made of the defense-in-depth for the sequence under consideration. As noted in RG 1.174 (Reference 6.3.2), it is not expected that there will be an equal apportionment of capabilities from one defense-in-depth criterion to another, only a "reasonable balance." Thus, allowance is made in the methodology for one of the criteria to be valued as Low Defense-in-Depth, and still have a positive overall assessment if other criteria compensate.

For the overall attribute evaluation-based assessment, sequence defense-in-depth is determined to substantiate the low frequency in the plant-specific PRA for the sequence, and thus, to be consistent with the defense-in-depth philosophy, if:

- none of the five criteria is valued as Low Defense-in-Depth, or

- no more than one criterion is valued as Low Defense-in-Depth, with at least two other criteria valued as High Defense-in-Depth

This attribute evaluation-based assessment is then used to either screen the sequence out of, or screen the sequence in to, the EPZ technical basis, consistent with the defense-in-depth philosophy.

3.5.3   Comparison of Defense-in-Depth Methodology to RG 1.174-Based Approach

To provide confidence in NuScale's methodology to evaluate defense-in-depth, a comparison to the layers and factors from RG 1.174 (Reference 6.3.2) is performed.

The four RG 1.174 layers of defense (i.e., successive measures) to protect the public are as follows:

Layer 1: Robust plant design to survive hazards and minimize challenges that could result in an event occurring

Layer 2: Prevention of a severe accident (core damage) should an event occur

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

TR-0915-17772-NP
Rev. 1

Layer 3: Containment of the source term should a severe accident occur

Layer 4: Protection of the public from any releases of radioactive material (e.g., through siting in low population areas and the ability to shelter or evacuate people, if necessary)

It can be seen from a comparison of these four layers with the five criteria from Section 3.5.2 that Layer 1 envelopes Criterion 1 and Criterion 2, and that Layers 2, 3, and 4 correspond to Criteria 3, 4, and 5.

RG 1.174 goes on to state the NRC finds it acceptable for a licensee to use seven factors to evaluate how a proposed licensing change impacts defense-in-depth. These are:

1. Preserve a reasonable balance among the layers of defense.

2. Preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures.

3. Preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty.

4. Preserve adequate defense against potential common-cause failures (CCFs).

5. Maintain multiple fission product barriers.

6. Preserve sufficient defense against human errors.

7. Continue to meet the intent of the plant's design criteria.

The purpose of the defense-in-depth assessment proposed in the NuScale EPZ size determination is to confirm screening of individual core damage accident sequences for possible use in determining an appropriate EPZ size. Assessing defense-in-depth on an individual sequence basis requires a different process compared to that used to assess defense-in-depth for a complete plant design, and will be different from that used to assess potential changes to an already licensed design. Nevertheless, it is possible to draw a parallel between the NuScale EPZ defense-in-depth approach and that suggested in RG 1.174. In particular, it can be seen that the method used in the NuScale EPZ addresses the seven factors identified.

Factor 1 (balance among the layers of defense) – In the NuScale method, defense-in-depth is characterized by five levels. In order for a core damage sequence to be screened-out of the NuScale EPZ size assessment, at least four of the five levels must be assessed as either medium or high. At most, only one of the five levels can be determined to be low and result in the sequence being screened-out from further consideration. Preservation of balance among the layers of defense is, therefore, inherent in the NuScale method.

Factor 2 (no overreliance on programmatic activities) – In order for a layer of defense to be assessed as medium or high, that attribute (using the term employed in the NuScale

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

TR-0915-17772-NP
Rev. 1

approach) needs to be satisfied by automatic plant response to an upset condition. In most cases, a high rating can only be achieved if the plant responds passively. If manual actions are required, that attribute is typically assessed as low. In no case can an attribute be assessed as high if it requires any type of human engagement at the time of the demand. This is characterized either explicitly or implicitly through the use of quantitative acceptance thresholds. A review of the assessments of the NuScale defense-in-depth criteria reveals minimal reliance on programmatic activities.

Factor 3 (system redundancy, independence, and diversity) – Many of the evaluations of the five criteria in the NuScale method rely on quantitative assessments of system failure probabilities. The quantitative acceptance thresholds are direct reflections of the amount of redundancy, independence, and diversity that characterize those aspects of the design directly associated with the particular core damage sequence being screened. This factor is therefore quantitatively captured in the NuScale method.

Factor 4 (potential CCFs) – As described above for Factor 3, this factor is also reflected in the quantitative thresholds used in the NuScale method. It is noted here that the quantitative PRA results used in the assessment of defense-in-depth in the NuScale method are extracted from a PRA that has already been reviewed by the NRC staff (either for DCA or for COL application reviews). This provides reasonable confidence in the robustness of the PRA results and a much more reliable basis for assessing defense-in-depth in contrast to a purely qualitative and subjective approach.

Factor 5 (multiple fission product barriers) – Although this design characteristic is inherently enhanced in the NuScale design (i.e., a relatively small core contained in the RPV, which in-turn is contained in the containment vessel (CNV), which in-turn is housed in the seismic Class-1 RXB), the NuScale method for assessing defense-in-depth for EPZ size addresses this factor through the assessment of the potential for an individual core damage sequence to result in a large release. This factor is therefore addressed in the NuScale method through the use of quantitative acceptance thresholds.

Factor 6 (human errors) – This factor has some overlap with the factor on programmatic activities (Factor 2). As such, it is also covered in the NuScale method by virtue of typically equating reliance on human actions with a low level of defense-in-depth. Additionally, the quantitative PRA results used for multiple criteria implicitly include any contribution from human errors. Therefore, this factor is included in the NuScale method.

Factor 7 (plant's design basis) – This factor is a fundamental consideration in the NuScale method. In multiple criteria, the response of safety systems to the individual sequence is assessed. Additionally, the quantitative results from the "focused PRA" are used. In the focused PRA, only safety-related equipment is included in the quantitative assessment (i.e., only the plant's licensing design basis equipment is considered). Therefore, in the NuScale method, quantitative PRA results are used to characterize the plant's design basis for the purpose of assessing defense-in-depth on an individual core damage sequence basis.

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

TR-0915-17772-NP
Rev. 1

3.6   Seismic Event Screening

Seismic event risk is site-dependent and impacts all plant structures, systems, and components (SSCs). In contrast with internal PRA initiators and other external events, seismic sequences may be evaluated by an SMA. A PRA-based SMA models seismic structural failures and seismically-induced initiators by using similar event tree logic as is used for internal events. However, the hazard frequency is not quantified. Instead, seismic risk is evaluated in terms of likelihood of core damage associated with a given peak ground acceleration (PGA).

Individual seismic failures are evaluated by calculating the failure probability for a given PGA using the median capacity and uncertainty parameters of the SSC. The characteristic fragility parameter HCLPF is also derived from the same parameters. Seismic results are quantified using the MIN-MAX[3] method as applied to accident sequences, which gives the controlling seismic failure associated with each cutset following a seismic event. This means that the HCLPF for an accident sequence is determined by the most limiting component.

The plant-level HCLPF corresponds to the core damage sequence with the lowest HCLPF. In accordance with Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application, ISG-020 (Reference 6.3.10), the acceptability threshold for an SMA is a plant-level HCLPF of 1.67 times the SSE of the plant design. For a NuScale SSE of 0.5g, this equates to a 0.84g HCLPF requirement.

This threshold of 1.67 x SSE is to be used as the screening criterion for EPZ sizing for seismic sequences. COL applicants shall update the design-specific plant system and accident sequence analysis to incorporate site-specific effects (e.g., soil liquefaction, slope failure) and plant-specific features (safety-related site-specific structures), as applicable, and screen each seismic sequence by the 1.67 x SSE criteria. Since the plant-level HCLPF is equal to the lowest individual sequence HCLPF, this is equivalent to demonstrating that the design-specific plant-level HCLPF capacity of 1.67 x SSE is maintained in the COL application.

HCLPF screening is being used rather than seismic PRA sequence frequency screening for the following reasons:

- A seismic PRA is not required as part of design certification, nor is it required as part of ESP or COL applications. However, if the site-specific SMA does not pass HCLPF screening, the COL applicant will demonstrate the acceptability of seismic risk, either by performing a seismic PRA or making design modifications.

---

[3] The MIN-MAX method is a process for sorting seismic failures corresponding to a particular ground motion (see Reference 6.3.12). Within a cutset, the highest HCLPF value is the controlling seismic failure. Between cutsets (or sequences), the cutset with the lowest HCLPF value is the controlling cutset.

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

TR-0915-17772-NP
Rev. 1

- A plant-level HCLPF applies to a severe accident sequence (i.e., 95 percent confidence that the plant will not experience core damage or large release with probability greater than 0.05 at this ground motion). The HCLPF represents actual seismic failures and their consequences, rather than simple exceedance of design parameters for a given SSC. Seismic failure probabilities would increase with ground motions above 1.67 x SSE, but the dominant failures are not expected to change. Thus SMA results and plant-level HCLPF already encompass any seismic risk insights related to beyond-design-basis earthquakes.

- The annual frequency of exceedance on a typical site hazard curve decreases by approximately an order of magnitude between 0.5g and 0.84g according to Safety/Risk Assessment Results for Generic Issue 199 (Reference 6.3.11). Beyond this range, decreases in exceedance frequencies correspond to very high ground motions (>1g) and higher associated uncertainties for most sites. Seismic risk then becomes dominated by rare, severe earthquakes that are less sensitive to design engineering and emergency planning considerations.

- In the event of a large, low likelihood earthquake, the emergency plans anticipated for a NuScale plant are expected to be as effective as a traditional nuclear plant emergency plan with a 10-mile EPZ radius. Regardless of EPZ distance, NuScale will have an on-site emergency plan to respond to a severe event. Additionally, it is expected that local government jurisdictions will develop plans, such as an integrated, all-hazards off-site plan that, although not subject to NRC or FEMA review, would function to mobilize response and prioritize allocation of resources. Taken together, such plans would be able to respond to the infrastructure damage and societal risks which would be much greater than the accident risks from the NuScale plant.

- Post-accident mitigation inside the EPZ is dependent on the surrounding infrastructure such as roads and bridges, which are likely to be damaged by an earthquake with a PGA of 1.67 x SSE or higher.

- Since the NuScale seismic margin is dominated by structural failures, engineering the seismic capacity is limited by the physical properties of the structural materials such as concrete and steel. Consequently, additional defense-in-depth features are likely to possess the same material vulnerabilities to seismic forces and would not significantly reduce the probability of core damage.

The 1.67 x SSE screening criterion is applied to each sequence that contains a seismic initiator. A sequence is only screened in for EPZ consideration if the limiting component fails at or below 1.67 x SSE and no design modification is performed. In that case, a seismic PRA would be performed, and the CDF for the sequences that do not pass the HCLPF criteria should be used.

3.7 Multi-Module Accident Methodology

This section describes the methodology for assessing physical interactions between modules under severe accident conditions for the NuScale plant. Mechanisms that can cause damage to a single module are analyzed in terms of their potential to propagate to other modules. Only sequences that are screened in by the methodology in Section 3.4

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

TR-0915-17772-NP
Rev. 1

for a single module are subjected to additional multi-module screening. This is because an accident sequence needs be credible in a single module to be credible for multiple modules.

Appendix D contains an example assessment of the potential simultaneous or near-simultaneous effects of all initiators on multiple modules by qualitatively discussing factors that contribute to, or limit, correlation between modules.

Several aspects of accident sequences have potential multi-module effects. The following aspects are considered in the assessment process shown in Figure 3-4:

- initiating events – initiators may affect multiple modules by definition, such as loss of off-site power (LOOP) 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. For example, the emergency core cooling system (ECCS) is initiated in all 12 modules following a loss of DC power, but multiple ECCS failures would not be correlated by the initiating event.

- human actions – actions performed on separate modules can have a degree of dependency. Several NuScale systems are shared in whole or in part between multiple modules, consequently the results of the associated human actions are dependent on shared equipment availability and available operator resources.

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure 3-4. Multi-module assessment process

Section 3.7.1 describes the multi-module implications associated with each type of initiator, including during low power and shutdown modes. Section 3.7.2 assesses correlated failures between modules. Section 3.7.3 assesses shared system failures. These sections detail the specific considerations that will be evaluated to follow the process shown in Figure 3-4. Additionally, Appendix D contains an example of qualitative assessment following this methodology.

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

TR-0915-17772-NP
Rev. 1

3.7.1   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 3-4. These hazards have the potential to include the following:

- internal events

- internal fires

- internal floods

- high winds

- external floods

- seismic events

Depending on which sequences screen in, the assessment may include both the full power and low power and shutdown operation modes.

Potential design-specific multi-module implications and coupling mechanisms are identified. Design-specific multi-module implication 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.

3.7.1.1 Internal Events

The assessment of internal event initiators will identify the originating system and component failure. If the system is shared between multiple modules, multi-module effects will be considered further. Additionally, if the initiator directly affects the safe shutdown response of more than one module, it will also be considered. Examples of internal event screening assessments are shown in Table D-1.

3.7.1.2 Low Power and Shutdown Modes

Low power and shutdown configurations present risks for multi-module accidents because of the module transport and refueling operations. While only one module can be transported or refueled at a time, it passes in physical proximity to other modules during the refueling process and is detached from its normal supports.

In accordance with the multi-module screening process shown in Figure 3-4, if there are any accident sequences screened in that involve a low power and shutdown configuration, they will be examined for coupling mechanisms between modules. This includes physical interactions between modules, for which an example is shown in Appendix D. The module that is shut down for refueling may be counted in a potential multi-module accident. However, the reduced release potential and smaller radionuclide

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

TR-0915-17772-NP
Rev. 1

inventory of a shut down module should be taken into account when evaluating possible releases.

3.7.1.3 Internal Fires

Internal fires are events that initiate within the plant boundary and can propagate to one or more compartments. Internal fires can include wide-ranging effects on multi-module initiators as well as shared systems between modules. Possible fire-induced initiating events include the following:

- transients

- LOOP

- induced ECCS demand

- induced LOCA inside containment

A COL applicant will, therefore, consider the following when evaluating the multi-module risks of internal fire events:

- whether a fire initiator has the potential to affect both divisions of the equipment control system

- whether the fire can spread to multiple compartments, thereby defeating the physical separation between module-specific systems

- whether a fire initiator can affect system(s) shared by more than one module

Examples of fire-induced event screening assessments are shown in Table D-2.

3.7.1.4 Internal Floods

Similar to internal fire risk, internal flood risk is modeled using specific flood-induced failure mechanisms and mapping equipment in affected compartments.

In accordance with the approach outlined in Figure 3-4, an applicant will consider the following criteria that may present a multi-module interaction risk for a flooding sequence:

- whether the flood-induced initiator affects multiple modules simultaneously

- whether there is a NuScale design-specific vulnerability to mitigating function impairment for more than one module

3.7.1.5 High Winds

High winds have the potential to damage off-site power equipment, leading to a LOOP for all 12 modules. Both the alternate AC power system and backup diesel generators (BDGs) are susceptible to failure from high winds, which implies that extreme high wind events could lead to a prolonged loss of AC power. The impact of high winds on all off-

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

TR-0915-17772-NP
Rev. 1

site power equipment will be considered. Potential structural damage from high winds will also be considered; however, the expectation is that because the NuScale RXB is a Seismic Category I structure, it is not susceptible to damage from high winds, wind-generated missiles, or damage from other buildings.

3.7.1.6 External Floods

If there is a risk of external flooding, operators are expected to perform a controlled shutdown on all operating modules when thresholds are reached that indicate an external flood could affect plant systems or components. In the event that there is not sufficient warning, an external flood could result in a loss of all off-site power, which would affect all 12 modules. Multi-module consequences for external floods will, therefore, be examined in terms of shared systems and other coupling mechanisms between modules following an induced loss of all off-site power.

3.7.1.7 Seismic Events

Seismic events present a unique challenge because of their site-wide effects and hazard-specific failure modes. As discussed in Section 3.6, there is a unique screening criterion for seismic accident sequences based on HCLPF. However, if there are any seismic accident sequences that are screened-in, the following considerations will be evaluated:

- RXB structural failures that have the potential to affect multiple modules

- correlated seismic failures of identical components located in different reactor modules

Examples of seismic structural event screening are shown in Table D-3.

3.7.2 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 3-4, if the initiator can induce any correlated failures of independent systems, any multi-module impacts will be addressed.

3.7.3 Shared System Failures

Several NuScale plant systems are shared in whole or in part between multiple modules, meaning their failure could have multi-module impacts. In accordance with the process described in Figure 3-4, if the initiator either involves a shared system, or impacts a shared mitigating function between modules, any multi-module impacts will be addressed.

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

TR-0915-17772-NP
Rev. 1

3.7.4 Multi-Module Conclusions

3.7.4.1 Determination of Multi-Module Accidents

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

}}[2(a),(c)] This complete method, as shown in Figure 3-4, is used to fully consider multi-module impacts for EPZ and any multi-module sequences that screen in are included in the EPZ technical basis requiring 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.

3.7.4.2 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 units or modules failed. 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.

3.8 Other Risks

3.8.1 Spent Fuel Pool

Accidents involving the SFP have been eliminated from detailed consideration in the EPZ technical basis based on the justification in the following subsections.

3.8.1.1 Spent Fuel Pool Boil-Off

Under this scenario, all 12 reactor modules are simultaneously shut down with no active cooling of the ultimate heat sink (UHS). The decay heat from the reactor modules and the 18-year inventory of spent fuel plus 13 fresh assemblies from the last fuel unload (i.e., maximum capacity of the spent fuel pool) 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 RXB. 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.

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

TR-0915-17772-NP
Rev. 1

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 indicate that the NuScale plant design provides significant time to heat up and boil down the SFP. These times are more than sufficient to take mitigating measures such as replenishing the SFP water inventory, thereby preventing fuel damage.

3.8.1.2 Criticality

The NuScale methodology for criticality analysis is applied to the fuel assemblies stored in the SFP. Particular consideration is given to 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 level 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.

3.8.1.3 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 is expected to be available to replenish the water inventory in the SFP (see Appendix E). Moreover, the ability to provide makeup water to the SFP with a Seismic Category I connection outside of the RXB has been included in the NuScale design.

3.8.2   Severe Accident Phenomena

Relevant severe accident phenomena, including in-vessel retention, fuel-coolant interaction, hydrogen combustion in containment, and high pressure melt ejection have been assessed for the NuScale design as part of the DCA. Each phenomena has been determined to either be not credible or does not pose a threat to containment integrity. As such, these severe accident phenomena are not considered further in the EPZ methodology.

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

TR-0915-17772-NP
Rev. 1

3.9 Security Events

Security events are addressed for completeness for EPZ.

3.9.1 Design-Basis Threat

NuScale designed the plant to address the regulatory requirements for the design-basis threat by incorporating security-by-design. 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. The COL applicant referencing the NuScale standard plant design will build upon the enhanced security features by developing a site-specific strategy to protect against radiological sabotage, as outlined in Purpose and Scope, 10 CFR 73.1 (Reference 6.3.15).

3.9.2 Beyond-Design-Basis Events

3.9.2.1 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 6.3.14). 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.

NuScale has performed structural and heat removal analyses, which substantiate that the RXB meets the aforementioned regulatory requirements. The NuScale RXB is an aircraft impact resistant structure.

3.9.2.2 Loss of Large Area

Conditions of Licenses, 10 CFR 50.54, Section (hh)(2) (Reference 6.3.16), requires that new plants constructed and operated in the United States consider the loss of large areas (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:

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

TR-0915-17772-NP
Rev. 1

1. Firefighting

2. Operations to mitigate fuel damage

3. Actions to minimize radiological release

NuScale has performed an analysis for LOLA using the guidance provided in B.5.b Phase 2 & 3 Submittal Guidance, NEI 06-12 (Reference 6.3.17) to demonstrate compliance with LOLA regulatory requirements.

3.9.3 Conclusions

A review of security events was conducted to determine potential impact on the EPZ technical basis.

NuScale has incorporated security-by-design in the standard design. The COL applicant will develop a site-specific protective strategy to protect against radiological sabotage caused by the design-basis threat.

An assessment of the effects on the NuScale design of the impact of a large commercial aircraft on the ability to maintain cooling of fuel in the reactor and SFP was performed. The insights gained from the performance of the aircraft impact assessment provided inputs to designing the RXB to meet this regulatory requirement.

An assessment of the effects on the NuScale design of a LOLA event was performed. COL applicants implementing the NuScale strategy outlined in the assessment will meet this regulatory requirement.

It is, therefore, concluded that security, aircraft impact, and LOLA events do not require further consideration in the EPZ technical basis, as all requirements have already been shown to be met.

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

TR-0915-17772-NP
Rev. 1

4.0   Design-Specific Methodology for Source Term and Dose Evaluations

4.1   Application of Software

The plume exposure EPZ size for a NuScale plant will reflect the design, source terms, and severe accident dose characteristics. The MELCOR and NRELAP5 codes are recommended for a user of 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 which has been NRC-developed for dose calculations, is required for use in this methodology. For all codes used, it is recommended that the latest final and approved version released to all users by the developer of each code be used. If another version is released during the analyses, it should be confirmed that the EPZ technical basis does not change with the newer version.

The NRELAP5 and MELCOR computer codes are used in the example severe accident analyses. The NRELAP5 code is the NuScale proprietary 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., NRELAP5) 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. 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 required severe accident software that must be used to comply with the NuScale EPZ sizing methodology. However, the primary code used to determine source terms will have the ability to evaluate the following severe accident phenomena:

- the thermal hydraulic response in the RCS, containment, and confinement buildings.

- core heat up, degradation, and relocation.

- hydrogen production, transport, and combustion.

- fission product release and transport behavior.

The use of a secondary code to confirm the thermal hydraulics up to core damage is not required as part of this methodology. However, the user of the methodology needs to have reasonable confidence in the severe accident results. Confirmation of results up to core damage is one option to provide confidence in the severe accident results. Examples of confirmation calculations are provided in Appendix A and Appendix B for less severe and more severe accident sequences, respectively.

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

TR-0915-17772-NP
Rev. 1

To perform the accident source term evaluation, a design-specific NuScale integrated severe accident model will be developed. A reactor module model should have already been developed to support the site-specific PRA as part of the COL application. The model will contain, at a minimum, the RPV, CNV, important safety and nonsafety-systems, and associated control logic. Severe accident analyses are performed to help inform and evaluate the selected sequences to determine fuel/cladding failure with their concomitant radionuclide release fractions and timing. Fission product release is determined based on the amount of core damage resulting from the specific accident sequences determined to be part of the EPZ basis in Sections 3.4 and 3.5.

Unique NuScale design features can function to reduce the release of activity to the environment. Examples of such features are the RXB, the reactor pool, and deposition surfaces in piping prior to a pipe break.

A multi-compartment (control volume) NuScale RXB model should be developed so that fission product transport within the building can be analyzed and estimates can be provided of the hydrogen distribution in various RXB rooms. The NPMs are housed in the RXB, which is a structurally robust, highly-engineered, seismic Class 1 building. While it is not containment, it is reasonable and appropriate to take credit for accident mitigation in the building to determine a best estimate EPZ distance. The methodology for crediting RXB accident mitigation involves developing a multi-compartment RXB model (which may be a separate-effects model or an integrated model) with the following features:

- The multi-compartment RXB model should consider RXB rooms into which fission products, steam, and hydrogen could be introduced, adjoining rooms into which the fission products and gases could transport, and rooms which could exchange with the environment (including realistic building leakage and filtration).

- The reactor pool should be modeled to account for aerosol scrubbing in the event of a release below the pool level.

- The RXB model should consider the potential for hydrogen deflagration in rooms in which hydrogen is predicted to accumulate.

- Source terms for severe accident sequences that are determined to be appropriate for evaluation in the EPZ size basis should include the effect of operationally-focused mitigation capabilities (as discussed in Appendix E) including crediting RXB accident mitigation systems.

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 RXB model. Separate-effects containment bypass piping models may be developed for unisolated chemical and volume control system (CVCS) outside containment break simulations (as in Appendix B) and other bypass piping pathways (such as the steam generator tubes). The CVCS piping has significant surface area for deposition of fission product aerosols released from the core.

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

TR-0915-17772-NP
Rev. 1

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 [References 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 RCS, 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 the secondary code that may optionally be used to confirm thermal hydraulic results in the primary code up to core damage is the NRELAP5 code, which is the reference example used throughout this report. NRELAP5 is NuScale's proprietary system transient thermal hydraulics code. NRELAP can only be used to model events up to the time of core damage because it doesn't have models for post core damage behavior. A design-specific NuScale model should be developed in NRELAP5 (or another code) for the purpose of modeling and 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 NuScale plant for confidence in the primary severe accident code.

4.1.2   Off-site Consequence Software and Modeling

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.

Depending on the sequence, the input to MACCS will describe the source term associated with core damage in a single (or multiple) NPM(s) within the RXB 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 movement or evacuation. The exposure durations used

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

TR-0915-17772-NP
Rev. 1

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 new 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. 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 most recent dose conversion factor (DCF) file included with the latest released version of MACCS should be used and 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). 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. 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,

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

TR-0915-17772-NP
Rev. 1

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 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 atmospheric transport and dispersion 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 a NuScale plant (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 NuScale design-specific 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. To aid in overall understanding of Section 4.0, Table 4-1 provides a summary of the NuScale 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 NuScale design-specific source term and dose methodology for developing EPZ size basis

Accident Type	Source Term Evaluation	Dose Evaluation
DBA (Section 3.3)	**Required:** • Apply DBST fission product release from containment for off-site dose calculated for Chapter 15 • No credit for RXB	• Apply MACCS dose-in-place evaluation (different from Chapter 15 dose evaluation) • Confirm or adjust MACCS modeling inside 0.5 km • Dose Criterion a (Section 3.2) with exposure duration of 96 hours
Less severe accidents (Section 3.4)	**Required:** • Apply full module severe accident model to calculate fission product release from the module due to containment leakage **Optional:**	• Apply MACCS dose-in-place evaluation • Confirm or adjust MACCS modeling inside 0.5 km • Dose Criterion b (Section 3.2) with exposure duration of 96

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

TR-0915-17772-NP
Rev. 1

Accident Type	Source Term Evaluation	Dose Evaluation
	• Apply RXB model to calculate fission product holdup and deposition, and fission product release to environment (including effects of realistic RXB air exchange rate) • Consider all operator mitigation actions (e.g., RXB sprays)	hours
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 • Apply RXB model to calculate fission product release from bypass piping to the RXB to calculate fission product holdup and deposition, and fission product release to environment (including effects of realistic RXB air exchange rate including filtration) • Consider all operator mitigation actions (e.g., RXB sprays)	• Apply MACCS dose-in-place evaluation • Confirm or adjust MACCS modeling inside 0.5 km • Dose Criterion c (Section 3.2) with exposure duration of 24 hours

4.2.1   Design-Basis Accidents

Source Term Evaluation Methodology

The accident sequence source term evaluation methodology for 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 NuScale FSAR (Reference 6.4.17). 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 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 bypasses the RXB and goes 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 NuScale design is

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

TR-0915-17772-NP
Rev. 1

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 Environmental Protection Agency (EPA) 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 recommended MACCS output parameter is ICRP60ED.

- When the wind shift without rotation plume model is used, which is the current best practice, peak dose[4] 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 azimuth 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.

- 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 MACCS results.

- 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

---

[4] 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.

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

TR-0915-17772-NP
Rev. 1

groundshine dose, and no shielding for cloudshine dose and 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 ICRP60ED dose exposure for 96 hours (345,600 seconds) 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	ICRP60ED	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

Section 3.4 addresses the methodology for determination of appropriate less severe accident sequences to be evaluated as part of the basis for NuScale 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. Example source term and dose calculations to illustrate the evaluation process for less severe accident sequences are provided in Appendix A. The example is intended to illustrate the methodology, but is not intended to be the basis for any NuScale design-specific plume exposure EPZ size.

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

TR-0915-17772-NP
Rev. 1

Source Term Evaluation Methodology

The methodology for the source term evaluation will apply the severe accident analysis thermal hydraulic software, with MELCOR and NRELAP5 used as examples. The methodology includes the following elements:

- Development and benchmarking of the primary severe accident (e.g., MELCOR) model for the NPM shall be performed on a NuScale design-specific basis 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 as discussed in Appendix E.

- The leakage of fission products from containment may be calculated by the severe accident code, and should be based on the assumption of a small opening in the top of containment (above the reactor pool surface so that no credit is taken for fission product scrubbing), where the opening is sized to result in design-basis leakage at containment design pressure that is commensurate with the technical specification limit for the NuScale CNV. 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 (as performed in Appendix A).

- A model of the RXB may optionally be used to credit fission product deposition and holdup in the building following leakage from the containment. The source term from the RXB 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

Section 3.4 also addresses the methodology for determination of more severe appropriate accident sequences to be evaluated as part of the basis for NuScale 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 (200 rem acute whole body probability of dose exceedance, see Section 3.2) is applied.

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

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

TR-0915-17772-NP
Rev. 1

illustrate the evaluation process for more severe accident sequences are provided in Appendix B. This example is intended to illustrate the methodology, but is not intended to be the basis for any NuScale design-specific plume exposure EPZ size.

Source Term Evaluation Methodology

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 RXB model, a separate effects model should be used to credit fission product deposition in the release pathway, such as piping, between the RPV and the RXB.

Dose Evaluation Methodology

The EPZ dose evaluation methodology for more severe accident sequences is based on Section 3.5 of the NEI white paper and applies Dose Criterion 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 at the distance where the probability of exceeding 200 rem whole body acute dose falls rapidly below 1E-3, consistent with the methodology and thresholds used in NUREG-0396.

  − 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

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

TR-0915-17772-NP
Rev. 1

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 24 hours 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 into the EPZ basis as determined in Section 3.4, the NUREG-0396 methodology for conditional probability of dose exceedance (conditional on core damage) versus distance will be applied. Specifically, a unique visual representation of conditional dose versus distance, such as Figure I-11 in NUREG-0396, will be created. This methodology is described in more detail in the subsequent paragraphs. However, based on the number of more severe sequences screened from Section 3.4, the resulting figure may look significantly different than Figure I-11, particularly if two or fewer accident sequences are screened-in, as the conditional probabilities would have much less impact. In this situation, an additional presentation of results for dose exceedance is recommended, which are individual sequence dose calculations compared to a 200 rem criterion. An example of this secondary presentation of results is found in Appendix B.

In equation form the probability of dose exceedance is as follows:

Consider a set of n more severe sequences, with core damage frequencies f_1, f_2, …,f_n. Let the total CDF from the PRA for all hazards, as determined in Section 3.4.1, be f_{total} (this total CDF should be approximately the same as the CDF of all screened in sequences as the most likely sequences are screened in). Let the conditional probability of dose exceedance (given core damage) for sequence i at distance j be p_{ij}. The conditional probability is determined over a set of weather trials. Then summed over all sequences, the conditional probability of dose D exceeding a given dose D_0 at distance j is:

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

TR-0915-17772-NP
Rev. 1

$$p_j(D > D_0) = \sum_{i=1}^{n} f_i p_{ij}/f_{total}$$
<div align="right">Equation 4-1</div>

A simple numerical example illustrating the steps for determining conditional probability of dose exceedance is as follows. Consider three sequences, S1, S2, and S3 with frequencies (CDF) as shown in Table 4-4 below, across the top. The total CDF is shown in the top right-hand cell. The conditional probability (given core damage) of dose exceeding 200 rem whole body acute for each of the three sequences is given for five distances from the reactor, 0.01 miles to 0.2 miles. The conditional probability of the dose exceeding 200 rem summed over all sequences at a given distance is in the right-hand column. From these values for the five distances in the numerical example, a curve similar to NUREG-0396, Figure I-11 (Reference 6.1.3) can be plotted and the distance at which probability drops below 1E-3 is determined, consistent with the methodology and thresholds used in NUREG-0396. In this example, the EPZ distance would be approximately 0.15 miles, as shown in Figure 4-1.

Table 4-4.   Example calculation of probability of dose exceedance

		Sequences			
		S1	S2	S3	Total CDF
	CDF	8.00E-06	5.00E-07	5.00E-08	8.55E-06
	Distance (mi)	**Cond. Prob. of exceeding 200 rem for sequence i at distance j**			**Total Cond. Prob. of exceeding 200 rem at distance j**
1	0.01	5.00E-02	7.00E-01	1.00E+00	9.36E-02
2	0.03	4.00E-02	7.50E-01	1.00E+00	8.71E-02
3	0.06	2.00E-02	6.00E-01	9.00E-01	5.91E-02
4	0.13	0.00E+00	8.00E-02	6.00E-02	5.03E-03
5	0.2	0.00E+00	0.00E+00	4.00E-04	2.34E-06

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

TR-0915-17772-NP
Rev. 1
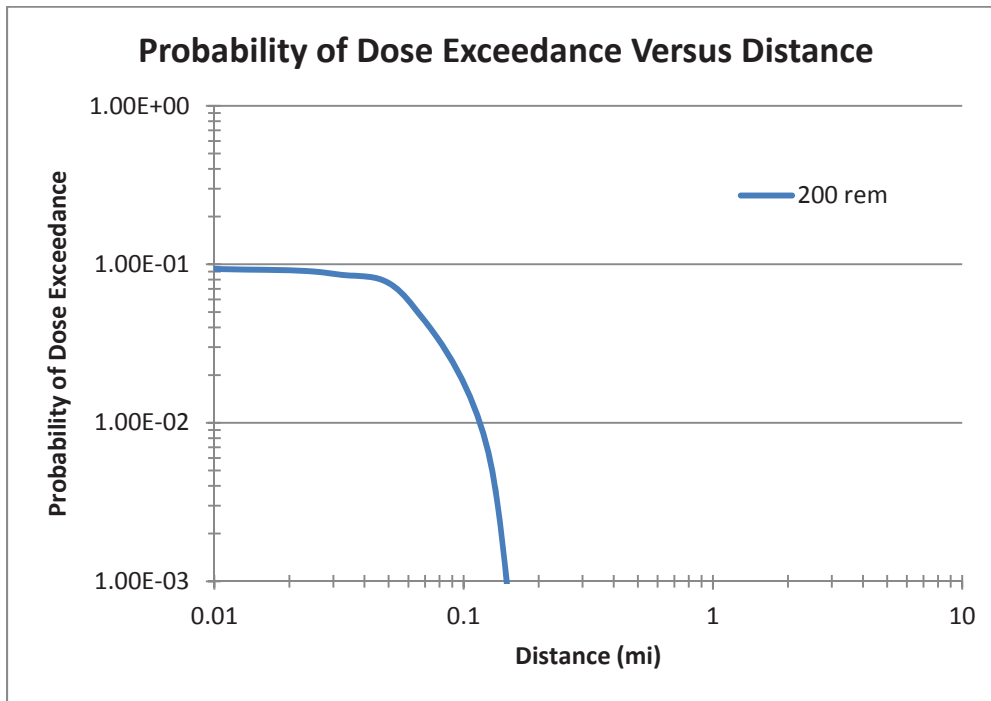
Figure 4-1. Example of NUREG-0396, Figure I-11 probability of dose exceedance curve based on Table 4-4 (Not actual results for the NuScale design)

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 complimentary cumulative distribution function output option in the WinMACCS EARLY output control parameter list. When the complimentary 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 over all weather trials considered. 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.

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

TR-0915-17772-NP
Rev. 1
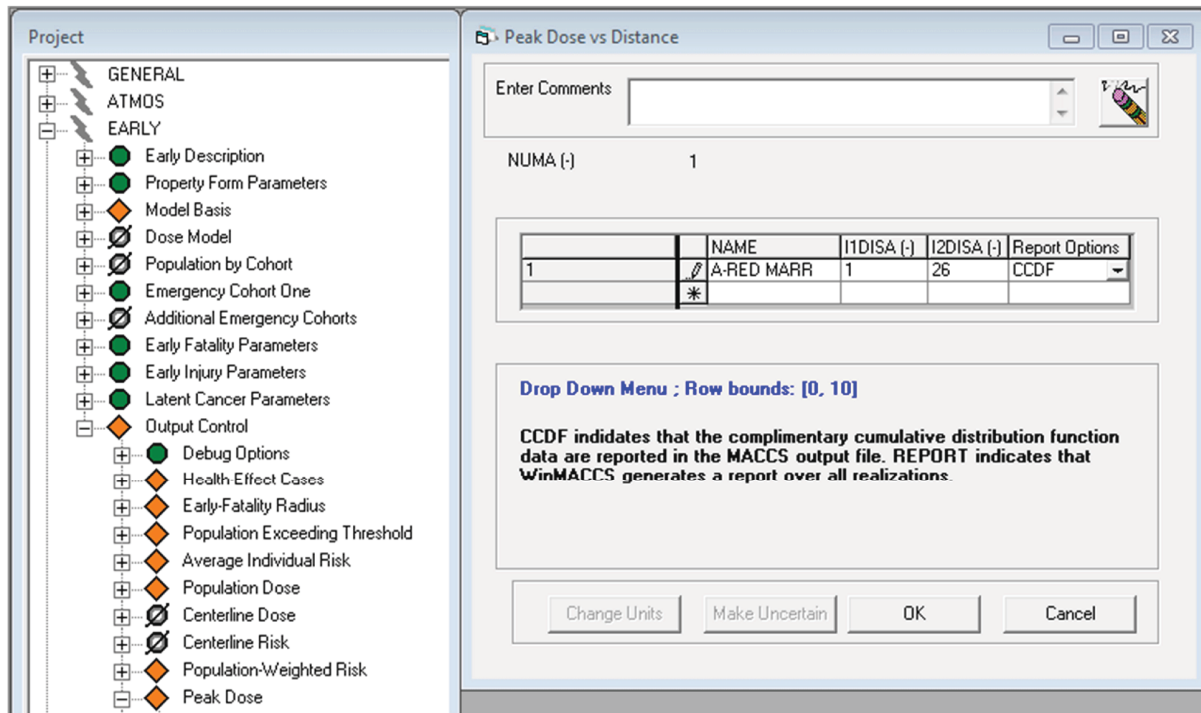
Figure 4-2. Example WinMACCS complimentary cumulative distribution function input selection

Figure 4-3. Example CCDF output for a single distance

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

TR-0915-17772-NP
Rev. 1

4.2.4 Additional Steps for Dose Evaluation Inside 0.5 Kilometers

With regard to calculating dose for distances at or near the site boundary, the MACCS user guide (Reference 6.4.3) cautions against the application of MACCS 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 the NuScale 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 atmospheric transport and dispersion (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 Chi/Q (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 from the NuScale RXB 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.

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

TR-0915-17772-NP
Rev. 1

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. 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 for the NuScale design, 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 and unrealistic accidents.

The framework for NuScale 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 methodology has the following general steps:

1. Selection of accident sequences

2. Identification of uncertainty parameters and definition of their distributions

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

TR-0915-17772-NP
Rev. 1

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

6. Confirmation of best-estimate values for important parameters

7. Repetition of source term and dose consequence analyses, if necessary

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 into the EPZ technical basis according to the methodology in Section 3.4. If there are no more 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 Step 2, 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 in light of the sequence selected. Both epistemic and aleatory parameters should be considered. The parameters from the SOARCA uncertainty analyses should be used as a starting point. Section C.2.3 contains an example of accident sequence and parameter selection.

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 NuScale 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 noninclusion.

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 [MARS]). 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.

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

TR-0915-17772-NP
Rev. 1

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 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.

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

TR-0915-17772-NP
Rev. 1

Uncertainty Analysis

Note: Curved boxes represent a transfer to or from the overall EPZ method.
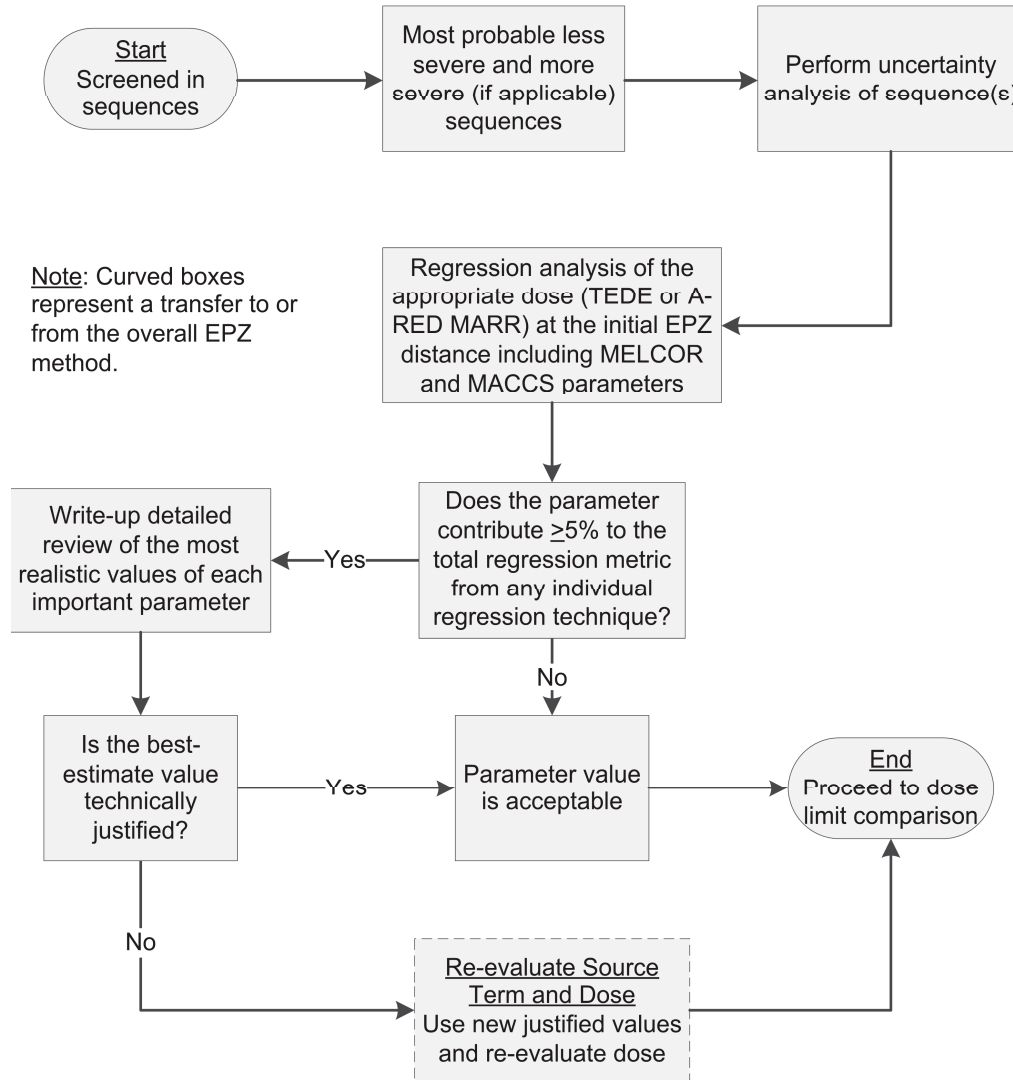
Figure 4-4. Application method of uncertainty results

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

TR-0915-17772-NP
Rev. 1

5.0   Summary and Conclusions on Methodology

The NuScale proposed approach for developing the technical basis for 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 that:

- it is design-specific, utilizing design-specific PRA information.

- 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 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 and a deterministic assessment of defense-in-depth in preventing and mitigating accidents that are addressed in the EPZ sizing methodology.

This LTR submits a proposed NuScale design-specific plume exposure EPZ sizing methodology for NRC review. NuScale requests, as part of this review and associated comment resolution, that the NRC provide an SER on the design-specific sizing methodology, including:

- a conclusion that the NuScale proposed plume exposure EPZ methodology in the LTR, when supported by design-specific information and appropriately implemented by the COL applicant, is an acceptable approach for determining the EPZ size for the NuScale design; and

- identification of any issues related to the NuScale EPZ technical basis that are to be resolved prior to or as part of the COL proceeding.

The design-specific 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 a NuScale plant. The methodology is applicable to any EPZ size, including the site boundary. The final EPZ size is the largest distance at which the dose consequence of each screened-in accident sequence is less than its respective dose criteria. Based on the results of applying the methodology, the final 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:

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

TR-0915-17772-NP
Rev. 1

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 design-specific, 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 design-specific PRA information as well as application of an INSAG-10 based approach (cross correlated to RG 1.174 guidance) developed by NuScale to assess accident sequence 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, which are unique to the NuScale design.

6. SFP accidents, severe accident phenomena, and security events have been addressed in the methodology with the conclusion that each does not require further consideration in the EPZ technical basis for the NuScale design.

7. A design-specific methodology for source term and dose evaluations has been defined that includes the appropriate application of software (such as NRELAP5, MELCOR, and MACCS) for the events identified. This evaluation includes additional steps to evaluate doses within 500 meters and an integrated uncertainty analysis methodology.

Example source term and dose analysis results for intact containment sequences and failed containment sequences are provided in Appendix A and Appendix B, respectively. Example PRA results and example assessment of accident sequence screening into the EPZ technical basis, including assessment of defense-in-depth, are provided in Appendix C. The example results indicate that accidents in the NuScale design are very infrequent and a site boundary EPZ is feasible since dose results do not exceed their respective criteria at the site boundary for the example accident sequences. An example of the application of the multi module evaluation methodology is presented in Appendix D and Appendix E provides an example methodology to credit operationally-focused mitigation capabilities. It should be noted that the results presented in Appendices A though E are solely intended to illustrate how the methodology would be implemented, and are not intended as the basis for a NuScale design-specific plume exposure EPZ size. NuScale is not requesting NRC approval of the examples in Appendices A through E as part of its review of this topical report.

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

TR-0915-17772-NP
Rev. 1

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).

6.1.3   U.S. Nuclear Regulatory Commission, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants," NUREG-0396/EPA 520/1-78-016, December 1978.

6.1.4   U.S. Environmental Protection Agency, "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," EPA-400/R-17/001, January 2017.

6.1.5   Nuclear Energy Institute, "Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone," December 23, 2013.

6.1.6   U.S. Nuclear Regulatory Commission, "Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors," SECY-11-0152, October 28, 2011.

6.1.7   U.S. Nuclear Regulatory Commission, "Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies," Staff Requirements Memorandum, SECY-15-0077, August 4, 2015.

6.1.8   U.S. Nuclear Regulatory Commission, "Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors," SECY-97-020, January 27, 1997.

6.1.9   *U.S. Code of Federal Regulations*, "Content of Applications; Technical Information," Section 34, Part 50, Chapter 1, Title 10, "Energy," (10 CFR 50.34(a)(1)).

6.1.10   U.S. Nuclear Regulatory Commission, "Rulemaking for Emergency Preparedness for Small Modular Reactors and Other New Technologies: Regulatory Basis," ADAMS Accession Number ML17206A265, September 2017.

6.2   Section 2.0 Document References

6.2.1   *U.S. Code of Federal Regulations*, "Combined Licenses," Subpart C, Part 52, Chapter 1, Title 10, "Energy," (10 CFR 52 Subpart C).

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

TR-0915-17772-NP
Rev. 1

6.2.2   *U.S. Code of Federal Regulations*, "Early Site Permits," Subpart A, Part 52, Chapter 1, Title 10, "Energy," (10 CFR 52 Subpart A).

6.2.3   U.S. Nuclear Regulatory Commission, "Options For Emergency Preparedness for Small Modular Reactors and Other New Technologies," SECY-15-0077, May 29, 2015.

6.2.4   *U.S. Code of Federal Regulations*, "Reactor Site Criteria," Part 100, Chapter 1, Title 10, "Energy," (10 CFR 100).

6.2.5   U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH 1400 (NUREG-75/014), October 1975.

6.2.6   U.S. Nuclear Regulatory Commission, "Request by Dominion Energy Kewaunee, Inc. for Exemptions from Certain Emergency Planning Requirements," SECY-14-0066, June 27, 2014.

6.2.7   U.S. Nuclear Regulatory Commission, "Request by Duke Energy Florida, Inc., for Exemptions from Certain Emergency Planning Requirements," SECY-14-0118, October 29, 2014.

6.2.8   *U.S. Code of Federal Regulations*, "Emergency Plans," Section 72.32, Part 72, Chapter 1, Title 10, "Energy," (10 CFR 72.32).

6.2.9   U.S. Nuclear Regulatory Commission, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants," NUREG-0654/FEMA-REP-1, Rev. 1, November 1980.

6.2.10 U.S. Nuclear Regulatory Commission, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs," SECY-10-0034, March 28, 2010.

6.2.11 U.S. Nuclear Regulatory Commission, "Criteria for Protective Action Recommendations for Severe Accidents," Volumes 1, 2, and 3, NUREG/CR-6953, December 2007.

6.2.12 U.S. Nuclear Regulatory Commission, "Identification and Analysis of Factors Affecting Emergency Evacuations," NUREG/CR-6864, SAND2004-5901, Volumes 1 and 2, January 2005.

6.2.13 U.S. Nuclear Regulatory Commission, "Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations," NUREG/CR-6981, October 2008.

6.2.14 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses (SOARCA) Report," NUREG-1935, November 2012.

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

TR-0915-17772-NP
Rev. 1

6.2.15 U.S. Nuclear Regulatory Commission, "Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies," SECY-16-0069, May 31, 2016.

6.2.16 U.S. Nuclear Regulatory Commission, "Rulemaking Plan on Emergency Preparedness for Small Modular Reactors and Other New Technologies," Staff Requirements Memorandum, SECY-16-0069, June 22, 2016.

6.3 Section 3.0 Document References

6.3.1 American Society of Mechanical Engineers/American Nuclear Society, "Standard for Level 1/Large Early Release Frequency Probabilistic Risk Assessment for Nuclear Power Plant Applications," Addendum A to RA-S-2008, ASME/ANS RA-Sa-2009, New York, NY.

6.3.2 U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis," Regulatory Guide 1.174, Rev. 3, January 2018.

6.3.3 U.S. Nuclear Regulatory Commission, "A Proposed Risk Management Regulatory Framework," A report to NRC Chairman Gregory B. Jaczko from the Risk Management Task Force, Commissioner George Apostolakis, Head, NUREG-2150, April 2012.

6.3.4 U.S. Nuclear Regulatory Commission, "Recommendations for Enhancing Reactor Safety in the 21st Century," NRC Near-Term Task Force Review of Insights from the Fukushima Dai-Ichi Accident, July 12, 2011.

6.3.5 U.S. Nuclear Regulatory Commission, "Risk Informing Emergency Preparedness Oversight: Evaluation of Emergency Action Level—A Pilot Study of Peach Bottom, Surry and Sequoyah," NUREG/CR-7154, January 2013.

6.3.6 U.S. Nuclear Regulatory Commission, "NRC Risk-Informed and Performance Based Initiatives," Presented by Commissioner George Apostolakis, American Nuclear Society Northeastern Section, Foxboro, MA, April 30, 2013.

6.3.7 U.S. Nuclear Regulatory Commission, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Regulatory Guide 1.183, July 2000.

6.3.8 U.S. Nuclear Regulatory Commission, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," NUREG-1860, Volumes 1-2, December 2007.

6.3.9 U.S. Nuclear Regulatory Commission, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application," Interim Staff Guidance, DC/COL-ISG-028, November 2016.

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

TR-0915-17772-NP
Rev. 1

6.3.10 U.S. Nuclear Regulatory Commission, "Interim Staff Guidance on Implementation of a Probabilistic Risk Assessment-Based Seismic Margin Analysis for New Reactors", Interim Staff Guidance, DC/COL-ISG-020, March 2010.

6.3.11 US Nuclear Regulatory Commission, "Safety/Risk Assessment Results for Generic Issue 199, 'Implications of Updated Probabilistic Seismic Hazard Estimates in Central and Eastern United States on Existing Plants'", GI-199, September 2010.

6.3.12 U.S. Nuclear Regulatory Commission, "An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Regulatory Guide 1.200, Rev. 2, March 2009.

6.3.13 International Atomic Energy Agency, "Defence in Depth in Nuclear Safety," INSAG-10, 1996.

6.3.14 *U.S. Code of Federal Regulations*, "Aircraft Impact Assessment," Section 50.150, Chapter 1, Title 10, "Energy," (10 CFR 50.150).

6.3.15 *U.S. Code of Federal Regulations*, "Purpose and Scope," Section 73.1, Chapter 1, Title 10, "Energy," (10 CFR 73.1).

6.3.16 *U.S. Code of Federal Regulations*, "Conditions of Licenses," Section 50.54, Chapter 1, Title 10, "Energy," (10 CFR 50.54).

6.3.17 Nuclear Energy Institute, "B.5.b Phase 2 & 3 Submittal Guideline," NEI 06-12, Rev. 3, July 2009.

6.3.18 U.S. Nuclear Regulatory Commission, "Standard Review Plan: Probabilistic Risk Assessment and Severe Accident Evaluation for New Reactors," NUREG-0800, Section 19.0, Rev. 3, December 2015.

6.3.19 U.S. Nuclear Regulatory Commission, "Functional Criteria for Emergency Response Facilities," NUREG-0696, February 1981.

6.4   Section 4.0 Document References

6.4.1 Gauntt, R.O., et al., "MELCOR Computer Code Manuals, Vol. 1: Primer and User's Guide," NUREG/CR-6119, Version 1.8.6, Rev. 3, Sandia National Laboratories, Albuquerque, NM, 2005.

6.4.2 Gauntt, R.O., et al., "MELCOR Computer Code Manuals, Vol. 2: Reference Manual," NUREG/CR-6119, Version 1.8.6," Rev. 3, Sandia National Laboratories, Albuquerque, NM, 2005.

6.4.3 U.S. Nuclear Regulatory Commission, "Code Manual for MACCS2 User's Guide," Vol. 1, NUREG/CR6613, SAND97-0594, May 1998.

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

TR-0915-17772-NP
Rev. 1

6.4.4   U.S. Nuclear Regulatory Commission, "MACCS Best Practices as Applied in the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," NUREG/CR-7009, August 2014.

6.4.5   U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project Volume 2: Surry Integrated Analysis," NUREG/CR-7110, Vol. 2, Rev. 1, August 2013.

6.4.6   U.S. Department of Energy, "MACCS2 Computer Code Application Guidance for Documented Safety Analysis, Final Report," Appendix C, DOE-EH-4.2.1.4-MACCS2-Code Guidance, June 2004.

6.4.7   U.S. Nuclear Regulatory Commission, "Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model," NUREG/CR-6853, October 2004.

6.4.8   U.S. Nuclear Regulatory Commission, "Atmospheric Relative Concentrations in Building Wakes," PNL-10521, NUREG/CR-6331, Rev. 1, May 1997.

6.4.9   U.S. Nuclear Regulatory Commission, "MELCOR Accident Consequence Code System (MACCS), Volume 2: Model Description," NUREG/CR-4691, February 1990.

6.4.10 U.S. Nuclear Regulatory Commission, "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," Draft NUREG, August 2007, ADAMS Accession Number ML072350221.

6.4.11 U.S. Nuclear Regulatory Commission, "MELCOR Best Practices as Applied to the State-of-the-Art Reactor Consequence Analyses (SOARCA) Project," NUREG/CR-7008, August 2014.

6.4.12 Electric Power Research Institute, "Modular Accident Analysis Program 5 (MAAP5) Applications Guidance: Desktop Reference for Using MAAP5 Software—Phase 3 Report," 3002010658, November 2017.

6.4.13 U.S. Nuclear Regulatory Commission, "RASCAL 4: Description of Models and Methods," NUREG-1940, December 2012.

6.4.14 Pacific Northwest National Laboratories, "Regional Atmospheric Transport Code for Hanford Emission Tracking (RATCHET)," PNWD-2224 HEDR, February 1994.

6.4.15 Lawrence Livermore National Laboratory, "Lagrangian Operational Dispersion Integrator (LODI) User's Guide," UCRL-AM-212798, June 2001.

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

TR-0915-17772-NP
Rev. 1

6.4.16 U.S. Nuclear Regulatory Commission, "State-of-the-Art Reactor Consequence Analyses Project: Uncertainty Analysis of the Unmitigated Long-Term Station Blackout of the Peach Bottom Atomic Power Station," NUREG/CR-7155, SAND2012-10702P, May 2016.

6.4.17 NuScale Power LLC, "NuScale Standard Plant Design Certification," January 2017, ADAMS Accession Number ML17013A229.

6.5   Section 5.0 Document References

There are no references in Section 5.0.

6.6   Appendices Document References

Appendix A Document References

6.6.1   U.S. Nuclear Regulatory Commission, "SOARCA Peach Bottom 2006 Meteorological File," April 2015, ADAMS Accession Number ML15097A114.

Appendix B Document References

There are no references in Appendix B.

Appendix C Document References

There are no references in Appendix C.

Appendix D Document References

There are no references in Appendix D.

Appendix E Document References

6.6.2   Control Room Staffing Plan Validation Results, RP-0516-49116 Rev. 1, ADAMS Accession Number ML16364A356.

6.6.3   Human Factors Engineering Operating Experience Review Results Summary Report, RP-0316-17614, Rev. 0, ADAMS Accession Number ML16364A342.

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

TR-0915-17772-NP
Rev. 1

Appendix A.   Example Source Term and Dose Evaluations – Less Severe Accidents

Note: The analysis of these accidents was performed prior to completion of the NuScale DCA,
thus the models used do not represent the final DCA design. However, as an example,
they are still representative of how the EPZ methodology would be used to perform
source term and dose evaluations.

A.1   Example Accidents to Be Evaluated

Section 3.4 of this report addressed the methodology for determination of appropriate
accident sequences (referred to as "accidents" throughout this appendix) to be evaluated
as part of the basis for NuScale EPZ size. To illustrate the methodology for less severe
intact containment accidents, three accidents were selected for source term and dose
evaluations in this appendix. The analyses of these accidents are solely meant to be an
example of application of NuScale's EPZ methodology. The three accidents are:

Loss of DC Power-2: Loss of DC power buses at time zero, both trains of decay
heat removal system (DHRS) actuate, ECCS fails, and containment isolates and
is intact

LOCA Inside Containment-4: LOCA inside containment other than a CVCS
injection line break, both trains of DHRS actuate, ECCS fails, and containment
isolates and is intact

CVCS LOCA Injection Line Inside Containment-4: LOCA in CVCS injection line
inside containment, both trains of DHRS actuate, containment isolates and is
intact

A.2   Example Evaluations for Time to Start of Core Damage

The example evaluations on time to start of core damage were performed to provide a
measure of confidence in the severe accident results; however, this type of evaluation is
not a requirement of the NuScale EPZ methodology. The example evaluations were
performed using a preliminary NuScale NRELAP5 model with the Beta 3.1 executable
version, in accordance with Section 4.2.2 methodology. For each of the above three
sequences, several variations were defined to provide completeness in the cases
considered. Table A-1 lists the variations that were considered.

Cases 1 and 2 are variations on loss of DC power-2 to address the two main ways that
ECCS can fail. Cases 3 through 8 are variations on LOCA inside containment-4. Cases
3, 4, and 5 are vapor break (reactor safety valve [RSV] opening at top of RPV) and
address the three ways that ECCS can fail. Cases 6, 7, and 8 are liquid breaks (in CVCS
discharge line) and address the three ways that ECCS can fail. Cases 9, 10, and 11 are
variations on CVCS LOCA injection line inside containment-4 and address the three
ways that ECCS can fail.

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

TR-0915-17772-NP
Rev. 1

Table A-1.   Accident variations calculated with NRELAP5

Case Number	Initiating Event	Case Identifier	DHRS*	RVV*	RRV*	RSV*	CNV Isolation
1	Loss of DC Power	LODC-01	2	2	0	0	Yes
2	Loss of DC Power	LODC-02	2	0	2	0	Yes
3	Spurious RSV Opening	SORSV-01	2	0	0	1	Yes
4	Spurious RSV Opening	SORSV-02	2	2	0	1	Yes
5	Spurious RSV Opening	SORSV-03	2	0	2	1	Yes
6	CVCS Liquid Discharge Break	LLC-01	2	0	0	0	Yes
7	CVCS Liquid Discharge Break	LLC-02	2	2	0	0	Yes
8	CVCS Liquid Discharge Break	LLC-03	2	0	2	0	Yes
9	CVCS Liquid Injection Break	LCC-01	2	0	0	0	Yes
10	CVCS Liquid Injection Break	LCC-02	2	2	0	0	Yes
11	CVCS Liquid Injection Break	LCC-03	2	0	2	0	Yes

*The "DHRS," "RVV," "RRV," and "RSV" column values in Table A-1 represent the number of actuated systems or components in each transient (spurious actuation in the case of the RSVs).

Core damage in the NRELAP5 runs is assumed to occur when peak clad temperature reaches 2200 F at which time the calculation is terminated. The NRELAP5 calculations (summarized in Table A-2, Figure A-1, Figure A-2, and Figure A-3) provide detailed thermal-hydraulic results as a function of time for peak cladding temperature (PCT), RPV collapsed liquid level, and CNV pressure, respectively. An interesting insight from Table A-2 is that, due to the low core linear generation rate and natural circulation core design, there is a significant time period between top of active core uncovery and time for a 2200 F PCT ({{   }}$^{2(a),(b),(c),ECI}$).

Table A-2.   Summary of NRELAP5 results on time to core damage

{{

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

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

TR-0915-17772-NP
Rev. 1

{{

}}[2(a),(b),(c),ECI]

*SORSV-01 did not reach core damage in 72 hours, although the water level above the top of active fuel (TAF) was decreasing and presumably would have eventually uncovered the core, eventually resulting in core damage.

**LCC-02 reached stable conditions at approximately 18.5 hours with water level never decreasing more than a foot below the TAF. The run was terminated at 21 hours.

{{

}}[2(a),(b),(c),ECI]

Figure A-1.   NRELAP5 plots of peak clad temperature versus time

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure A-2.   NRELAP5 plots of RPV level versus time

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure A-3. NRELAP5 plots of CNV pressure versus time

Summary observations from the NRELAP5 runs are as follows:

- Cases with all ECCS valves failing to open (Cases 3, 6, and 9), while relatively low in frequency since this is not a likely failure mode for ECCS, are shown for completeness. These cases do not reach core damage for {{ $\}\}^{2(a),(b),(c),ECI}$ or more hours since the rate of inventory loss from the RPV is so low.

- Cases with reactor vent valves (RVVs) open and reactor recirculation valves (RRVs) failing to open (Cases 1, 4, 7, and 10) reach core damage in the range of {{ $\}\}^{2(a),(b),(c),ECI}$ hours, somewhat faster than Cases 3, 6, and 9 due to the inventory loss (vapor flow) out of the RVVs with no flow path from the CNV back into the RPV.

- Cases with RVVs failing to open and RRVs open (Cases 2, 5, 8, and 11) reach core damage in the range of {{ $\}\}^{2(a),(b),(c),ECI}$ hours, somewhat faster than Cases 1, 4, 7, and 10 due to the higher rate of inventory loss from liquid flowing out of the open RRVs. General transient behavior observations common to all of these NRELAP5 calculations are:

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

TR-0915-17772-NP
Rev. 1

    – No increase in peak clad temperature occurs after any LOCAs until the core is uncovered later in the event. If there were no core uncovery, the PCT for these small break LOCAs would be the initial PCT at power operation.

    – Overall, these accidents progress slowly and there is significant time for EOP and operator mitigation actions which would prevent and mitigate core damage.

A.3 Example Source Term Evaluations

For the example source term evaluations for the less severe accidents, four of the sequence variations from Table A-2 were selected and MELCOR 1.8.6 calculations were performed using a preliminary NuScale MELCOR model with executable version 3964, in accordance with Section 4.2.2 methodology for these sequence variations:

- Case 1 (LODC-01) – Loss of DC power, both trains of DHRS actuate, RVVs open, RRVs fail to open, containment isolates

- Case 4 (SORSV-02) – Spurious opening of an RSV, both trains of DHRS actuate, RVVs open, RRVs fail to open, containment isolates

- Case 5 (SORSV-03) – Spurious opening of an RSV, both trains of DHRS actuate, RVVs fail to open, RRVs open, containment isolates

- Case 7 (LLC-02) – CVCS discharge LOCA, both trains of DHRS actuate, RVVs open, RRVs fail to open, containment isolates

The basis for selection of these four cases is that they provide a spectrum of the more likely intact containment accidents that result in core damage. The four cases include: a loss of DC power sequence (Case 1) as well as vapor (Cases 4 and 5) and liquid (Case 7) LOCAs; cases with ECCS failure where RVVs open, RRVs fail to open (i.e., {{ }}[2(a),(b),(c),ECI] to core damage, Cases 1, 4, and 7); and a case with ECCS failure where RVVs fail to open, RRVs open, which has the faster time to core damage (Case 5).

The MELCOR result of main interest for the example source term evaluations is the fission product release from containment as a function of time. That is, the release versus time as a fraction of core inventory for the important radionuclide groups that comprise the input to MACCS for the off-site dose calculation. As noted in Section 4.2 methodology, the containment release is assumed to be directly to the environment, and no credit is being taken for holdup or deposition of fission products in the RXB for these less severe example source term evaluations. The leakage from containment is conservatively modeled as 0.1 percent per day from the top of the containment (no pool scrubbing) with no reduction of this leakage over the 96-hour dose exposure duration of the accident. Chapter 15 of NuScale's FSAR (Reference 6.4.17) maximum hypothetical accident dose calculations typically allow for a 50 percent reduction in containment leak rate after 24 hours of a release based on the expected reduced containment pressure. The NuScale containment shows a large reduction in peak pressure for these cases

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

TR-0915-17772-NP
Rev. 1

before core damage occurs down to values of 20 percent or less of the containment design pressure of 1000 psia.

Figure A-4 and Figure A-5 show PCT and RPV collapsed liquid level, respectively, for the NRELAP5 calculations described in Section A.2 for the four cases compared with the NuScale MELCOR model calculations. The NRELAP5 and MELCOR simulation results show very similar accident progressions (Figure A-4) with similar mass transfer rates from the RPV to the CNV (Figure A-5). The two codes exhibit slight differences early in the simulations, MELCOR predicting about a {{ }}[2(a),(b),(c),ECI] shorter time to core damage when compared to NRELAP5. This difference is not considered to be significant given the different methodologies inherent in each of these computer codes (e.g., nodalization, conservation equations).

{{

}}[2(a),(b),(c),ECI]

Figure A-4. Peak cladding temperature, NRELAP5 versus MELCOR*

{{

}}[2(a),(b),(c),ECI]

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure A-5. RPV collapsed liquid level above TAF, NRELAP5 versus MELCOR

Figure A-6 shows containment pressure for the four cases. Containment pressure decreases rapidly shortly after accident initiation due to passive heat transfer from containment to the reactor pool, and the time of the beginning of the pressure transient correlates well with the time of start of core damage, the pressure increase being due to noncondensable gas generation.

{{

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

Figure A-6. Containment pressure versus time, MELCOR

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

TR-0915-17772-NP
Rev. 1

Figure A-7 shows the total inventories for the key radionuclide groups (aerosols) in the CNV (solid lines) plotted along with the respective deposited CNV inventories (dashed lines) for the loss of DC power case (Case 1). It is evident from Figure A-7 that essentially all of the aerosol deposits quickly (core damage in the MELCOR run starts at greater than {{ }}[2(a),(b),(c),ECI]), which is consistent with the small containment volume of a NuScale module. This enhances aerosol agglomeration and particle size growth, which increases sedimentation rate. Steam condensation-driven aerosol removal is also significant in the NuScale containment due to the passive heat transfer through the CNV walls to the reactor pool.

{{

}}[2(a),(b),(c),ECI]

Figure A-7. CNV radionuclide inventory for loss of DC power (Case 1), total and deposited

Figure A-8 through Figure A-11 show the MELCOR-calculated source terms from an intact containment to the environment for the key radionuclide groups for the four cases: loss of DC power with RVVs open, RRVs failing to open (Case 1); spurious opening of an RSV with RVVs open, RRVs failing to open (Case 4); spurious opening of an RSV with RVVs failing to open, RRVs open (Case 5); and CVCS discharge LOCA with RVVs open, RRVs failing to open (Case 7), respectively. The figure on the left is instantaneous release fraction (fraction of core inventory) at each time step, and the figure on the right is cumulative release fraction.

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure A-8. Loss of DC power (Case 1) – MELCOR-calculated source term to environment

{{

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

Figure A-9. Spurious opening of RSV with RVVs open, RRVs failing to open (Case 4) –
MELCOR-calculated source term to environment

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

TR-0915-17772-NP
Rev. 1

{{

}}[2(a),(b),(c),ECI]

Figure A-10. Spurious opening of RSV with RVVs failing to open, RRVs open (Case 5) – MELCOR-calculated source term to environment

{{

}}[2(a),(b),(c),ECI]

Figure A-11. CVCS discharge LOCA with RVVs open, RRVs failing to open (Case 7) – MELCOR-calculated source term to environment

Figure A-12 and Figure A-13 provide a comparison of the radionuclide release fractions for the two spurious RSV opening cases: Case 4 (RVVs open, RRVs failing to open) versus Case 5 (RVVs failing to open, RRVs open). Figure A-12 shows the cumulative radionuclide release fractions from the core into the containment, and Figure A-13 shows cumulative radionuclide release fractions from containment to the environment. The release fractions for Case 5 are about half of that of Case 4. The reason is that reflood of coolant through the open RRVs is predicted by MELCOR shortly after the start of core damage. The reflood occurs due to reduced volumetric flow of steam out the open RRVs as the core begins to uncover and increased static head of liquid in the CNV, which results in flow back into the RPV.

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

TR-0915-17772-NP
Rev. 1

This effect can be seen in Figure A-4, Figure A-5, and Figure A-6 at 7 to 8 hours into the accident, and results in termination of the accident and a smaller release compared to Case 4.

{{

}}2(a),(b),(c),ECI

Figure A-12. Comparison of release fraction from core to containment for the two cases of spurious opening of RSV (Case 4 versus Case 5)

{{

}}2(a),(b),(c),ECI

Figure A-13. Comparison of release fraction from containment to environment for the two cases of spurious opening of RSV* (Case 4 versus Case 5)

* It should be noted that to aid in visualization the x-axis is scaled relative to the time of initial release, meaning that time is shifted such that the first radionuclide release is at time 0 as opposed to the other figures, which have accident initiation at time 0.

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

TR-0915-17772-NP
Rev. 1

Summary observations from the MELCOR source term evaluations are as follows:

- The comparison of MELCOR with NRELAP5 calculated time to start of core damage in Figure A-4 and Figure A-5 shows close agreement. The following table presents a comparison of the MELCOR versus NRELAP5 time to core damage results for Cases 1, 4, 5, and 7.

Table A-3. Comparison of the MELCOR versus NRELAP5 time to core damage results for Cases 1, 4, 5, and 7

{{

}}[2(a),(b),(c),ECI]

- As shown in Figure A-7, fission product aerosols are removed rapidly in containment, which will contribute to smaller radionuclide release and lower off-site doses for the NuScale design.

- As noted, the rapid aerosol removal rate significantly reduces the amount of airborne aerosol available for release from containment.

- Figure A-12 and Figure A-13 show that while accidents with RVVs failing to open and RRVs open (such as Case 5) are faster to start of core damage, accident progression is terminated shortly after core damage starts and releases are even smaller than the accidents with RVVs open and RRVs failing to open (such as Case 4).

A.4 Example Dose Evaluations

For the example dose evaluations of less severe accidents, MACCS calculations were performed with Version 3.10.0, in accordance with Section 4.2.2 methodology for Cases 1, 4, 5, and 7 MELCOR-calculated source terms in Section A.3 above. A dose-in-place model was used (i.e., neither evacuation nor relocation outside the EPZ were credited in the exposure phase). The SOARCA Peach Bottom 2006 Meteorological File (Reference 6.6.1) was used for the dose evaluations on the basis that it was more bounding than the other meteorological files that were available. Shielding and protective factors used were {{ }}[2(a),(c)] The MACCS output reported for TEDE is ICRP60ED. The DCF file was modified from the original DCF file included with the release of Version 3.10.0 (FGR13DCF). This file contains factors from FGR-11 for inhaled exposure, FGR-12 for external exposure, and FGR-13 for cancer risks. The modification implemented SOARCA best practices with respect to residual cancer risk. However, cancer risk is not considered in this analysis and the DCFs for ICRP60ED are unchanged. As in earlier sections of Appendix A, the dose results presented here are illustrative only for the purpose of demonstrating the methodology and are not intended to be the basis for the NuScale design plume exposure EPZ size.

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

TR-0915-17772-NP
Rev. 1

A site boundary distance is not shown on Figure A-14 and Figure A-15, although doses at distances closer than 0.5 km were calculated using MACCS and are displayed. For licensing application, MACCS calculated doses for distances inside 0.5 km will be adjusted as necessary in accordance with Section 4.2.4.

{{

}}[2(a),(b),(c),ECI]

Figure A-14. Mean TEDE (rem) for Peach Bottom meteorology versus distance from reactor (EPZ size methodology limit is 1 rem TEDE)

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure A-15. 95th percentile TEDE (rem) for Peach Bottom meteorology versus distance from reactor (EPZ size methodology limit is 5 rem TEDE)

Figure A-14 and Figure A-15 show the mean and 95th percentile TEDE results, respectively, for the four source terms. Key results from the two figures are as follows:

- SORSV-02 is the highest dose source term and LODC-01 is the lowest, with the difference in dose generally less than a factor of three.

- For SORSV-02, mean TEDE at 0.5 km (~0.3 miles) is {{ $\}\}^{2(a),(b),(c),ECI}$ and 95th percentile TEDE is {{ $\}\}^{2(a),(b),(c),ECI}$.

Table A-3 summarizes doses for close-in distances for the largest of the four source terms (SORSV-02) and compares these doses with dose Criterion b from Section 3.2. Based on the dose evaluation, the TEDE for the less severe accidents has significant margin to the applicable dose criterion.

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

TR-0915-17772-NP
Rev. 1

Table A-4.   Comparison of doses for SORSV-02 (Case 4) with dose criterion

Distance (miles)	Mean TEDE (rem)		95% TEDE (rem)	
	Calculated	Criterion b	Calculated	Criterion b (Section 3.2)
1.25	{{ }}$^{2(a),(b),(c),ECI}$	1	{{ }}$^{2(a),(b),(c),ECI}$	5
0.8	{{ }}$^{2(a),(b),(c),ECI}$	1	{{ }}$^{2(a),(b),(c),ECI}$	5
0.3	{{ }}$^{2(a),(b),(c),ECI}$	1	{{ }}$^{2(a),(b),(c),ECI}$	5
0.1*	{{ }}$^{2(a),(b),(c),ECI}$	1	{{ }}$^{2(a),(b),(c),ECI}$	5

*Doses at this distance may need to be adjusted to address building wake effects as noted in Section 4.2.4.

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

TR-0915-17772-NP
Rev. 1

Appendix B.   Example Source Term and Dose Evaluations – More Severe Accidents

Note: The analysis of these accidents was performed prior to completion of the NuScale DCA,
thus the models used do not represent the final DCA design. However, as an example,
they are still representative of how the EPZ methodology would be used to perform
source term and dose evaluations.

B.1   Example Accidents to Be Evaluated

Section 3.4 describes the methodology for determination of appropriate accident
sequences (referred to as "accidents" throughout this appendix) to be evaluated as part
of the basis for NuScale EPZ size. As part of this methodology, screening is performed
for potential more severe, containment bypass accidents as depicted in Figure 3-3.
Those, if any, that are screened in will be considered as appropriate accidents to be
evaluated as part of the basis for NuScale EPZ size, including source term and dose
evaluations for comparison to dose Criterion c (See Section 3.2 of the main report). If a
given accident is not screened in, no source term and dose evaluation would be
performed as part of the EPZ size basis and the accidents would not be included in the
EPZ size basis.

There are two main classes of containment bypass sequences for full power internal
events: unisolated CVCS break outside of the CNV; and steam generator tube failure
(SGTF) with an unisolated secondary line break. To illustrate the methodology in Section
3.4, example screening is performed in Appendix C of all accidents, including
containment bypass accidents. The example applications in Appendix C indicated that,
based on the layers of defense-in-depth in the design and the low frequency, neither of
these containment bypass sequences were screened in to the EPZ technical basis for a
NuScale plant.

While the Appendix C example assessment of the two containment bypass accidents
indicates that these accidents should not be included in the EPZ basis, source term and
dose evaluations were performed for these accidents and on several sensitivity cases
based on these accidents. This was done not to justify EPZ size, but rather to illustrate
the source term and dose evaluation methodology.

The two sequences and associated sensitivity cases, and the basis for their selection,
are described below:

1. CVCS LOCA Injection Line Break Outside Containment

    Base Case: Break in CVCS injection line outside containment, both CVCS
    isolation valves fail to close, check valve fails to close, both trains of DHRS
    actuate, ECCS actuates, operator mitigation actions fail.

    Basis for Selection: This accident type has been historically important in severe
    accident work (e.g., interfacing system LOCA) and is a potentially important
    release path (containment bypass).

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

TR-0915-17772-NP
Rev. 1

Sensitivity Case A: Break in CVCS injection line outside containment, both CVCS isolation valves fail to close, check valve fails to close, both trains of DHRS actuate, no credit for ECCS, credit operator action for containment flooding via the containment flooding and drain system (CFDS) at 8 hours.

Basis for Selection: This sensitivity is to explore variations in the base case in order to get to core damage since the base case does not get to core damage (as discussed in Section B.2.1). Sequence frequencies are so low at this point (as can be seen in Table C-1 of Appendix C) that absolute probabilities are not a meaningful basis for selecting the sensitivity case. Thus, this specific equipment availability is determined by engineering judgment, not informed by the PRA, and on this basis, ECCS is not credited in the sensitivity case. Containment flooding via CFDS at 8 hours is conservatively selected based on the fact that operator action is considered very likely given:

- the simplicity of the action

- the time available and the indications that the operator would see (i.e., module and RXB conditions), and

- the expectation that, as noted in Appendix E, NuScale's operationally-focused mitigation capability includes containment flooding.

The 8 hours is also based on the SOARCA approach of crediting mitigation measures in 8 hours where there is potential for damage to the site outside of the RXB.

Sensitivity Case B: Break in CVCS injection line outside containment, both CVCS isolation valves fail to close, CVCS check valve fails to close, both trains of DHRS actuate, no credit for ECCS with operator action to initiate containment flooding using the CFDS at 24 hours.

Basis for Selection: This is to further explore temporal variations in the accident to get to core damage.

2. Pressure-Induced Steam Generator Tube Failure (Unisolated Secondary Line Break)

Base Case: Break in unisolable main stream line location, pressure-induced SGTF, one train of DHRS actuates (intact steam generator), ECCS actuates, and operator mitigation actions fail.

Basis for Selection: This accident type has been historically important in severe accident work (e.g., consequential steam generator tube failure) and is a potentially important release path (containment bypass).

As described below, time to core damage is evaluated for both of these sequences. The source term and dose evaluation was limited to the sensitivity cases for CVCS LOCA injection line outside containment since this pathway is expected to be the most

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

TR-0915-17772-NP
Rev. 1

consequential bypass event because of the larger size of the CVCS line as compared to a steam generator tube.

B.2   Example Calculations for Time to Start of Core Damage

B.2.1   Base Case Calculations with NRELAP

The example calculations on time to start of core damage for the two base case sequences were performed to provide a measure of confidence in the severe accident results; however, this type of evaluation is not a requirement of the NuScale EPZ methodology. The example calculations were performed using a preliminary NuScale NRELAP5 model with the Beta 3.1 executable version. Core damage in the NRELAP runs is assumed to occur when peak clad temperature reaches 2200 F. The NRELAP5 calculations were terminated when peak clad temperature reached 2200 F or at 72-hours, whichever occurred first.

CVCS LOCA Injection Line Outside Containment – Base Case

Figure B-1 through Figure B-6 provide the thermal-hydraulic results as a function of time for peak clad temperature, RVV and RRV flow rate, RPV collapsed liquid level, and CNV pressure.

The calculation was run to 72 hours and, as is evident from Figure B-1, no core damage occurs. This figure shows that the PCT of this sequence occurs at full power operation before the initiation of the CVCS pipe break. From Figure B-2 and Figure B-3, ECCS actuates on low RPV water level at about 8 minutes into the accident, and from Figure B-4, at about 5 hours, the RPV collapsed level stabilizes at ~3 feet above the core. RPV water level remains at this elevation for the duration of the accident out to 72-hours. This steady-state cooling is expected to continue much longer. The reason for this stable situation is that the system reaches mass and thermal equilibrium after several hours, and there is little or no coolant inventory being lost out the break. From Figure B-5 and Figure B-6, the pressure inside the module equilibrates with the outside pressure (atmospheric) after about 4 hours. Decay heat is being removed by natural circulation through the ECCS (steam flowing out the RVVs and condensed liquid flowing back in through the RRVs) and heat transfer through the CNV wall to the reactor pool. The CNV pressure drops to below 50 psia in less than 3 hours due to the passive heat removal into the UHS.

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-1. Peak clad temperature – Base Case CVCS LOCA Outside Containment (72 Hours)

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-2. ECCS flow rates – Base Case CVCS LOCA Outside Containment (72 Hours)[5]

---

[5] There are instances as presented, where results corresponding to multiple cases may overlap and thus only one line is visible. The RVV_2 trend line overlaps the RVV_1 trend line and the RRV_2 trend line overlaps the RRV_1 trend line.

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-3. ECCS flow rates – Base Case CVCS LOCA Outside Containment (5000 Seconds)[5]

{{

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

Figure B-4. RPV collapsed level – Base Case CVCS LOCA Outside Containment (72 Hours)

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-5. Containment pressure – Base Case CVCS LOCA Injection Containment (72 Hours)

{{

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

Figure B-6. Containment pressure (expanded scale) – Base Case CVCS LOCA Injection Containment

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

TR-0915-17772-NP
Rev. 1

<u>Pressure-Induced Steam Generator Tube Failure (Unisolated Secondary Line Break) –
Base Case</u>

This calculation was also run to 72 hours. Figure B-7 through Figure B-12 provide the
thermal-hydraulic results as a function of time for peak clad temperature, RVV and RRV
flow rate, RPV collapsed liquid level, and CNV pressure.

As shown in Figure B-7, no core damage occurs. As in the case of the previous CVCS
injection line break case, maximum PCT is the full power operation value prior to the
SGTF. From Figure B-8 and Figure B-9, ECCS actuates on low RPV water level just
beyond 30 minutes into the accident, and, from Figure B-10, at about 5 hours the RPV
collapsed level stabilizes at ~6 feet above the core. RPV water level remains at this
elevation for the duration of the accident out to 72 hours, and this steady state cooling is
expected to continue much longer. Similar to the CVCS outside break, the reason for this
stable situation is that the primary coolant system reaches pressure equilibrium with
atmospheric pressure after several hours resulting in little or no coolant inventory being
lost out the path through the failed steam generator tube and broken secondary line.
From Figure B-11 and Figure B-12, the pressure inside the module equilibrates with the
outside pressure (atmospheric) after about 6 hours. Decay heat is being removed by
natural circulation through the ECCS (steam flowing out the RVVs and condensed liquid
flowing back in through the RRVs) and heat transfer through the CNV wall to the UHS.

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-7. Peak clad temperature – Base Case Pressure-Induced SGTF (72 Hours)

{{

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

Figure B-8. ECCS flow rates – Base Case Pressure-Induced SGTF (72 Hours)[5]

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-9. ECCS flow rates – Base Case Pressure-Induced SGTF (5000 Seconds)[5]

{{

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

Figure B-10. RPV collapsed level – Base Case Pressure-Induced SGTF (72 Hours)

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-11. Containment pressure – Base Case Pressure-Induced SGTF (72 Hours)

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-12. Containment pressure (expanded scale) – Base Case Pressure-Induced SGTF

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

TR-0915-17772-NP
Rev. 1

Observation from Base Case NRELAP Runs

The important observation from the base case NRELAP runs is that core damage does not occur by the end of the calculation time of 72 hours and is not expected to occur at all for the base case accidents where ECCS actuates. This is because the system reaches a pressure and temperature equilibrium state after several hours with several feet of water above the core and where there is little or no further coolant inventory being lost outside containment through the break. Decay heat is being removed passively via the ECCS through the CNV wall to the UHS. The CNV pressure drops to less than 50 psia in less than 3 hours due to this passive heat removal mechanism to the UHS.

B.2.2   CVCS Outside Break Test Case Calculation

Neither of the base cases that were evaluated in Section B.2.1 reached core damage. The two sensitivity cases defined in Section B.1 are intended to explore getting to core damage for the CVCS outside break by taking no credit for ECCS, but considering other mitigation actions which are likely given the length of time available to the operators.

Before evaluating the two sensitivity cases (discussed in Section B.3), which used the NuScale MELCOR model as discussed in Section B.3, a test case calculation was performed for the CVCS outside break with no credit for ECCS and no credit for any operator mitigation actions. The test case compares NRELAP5 and the NuScale MELCOR model. It is being applied here to: (1) provide a benchmark for time to start of core damage for MELCOR; and (2) confirm that there is sufficient time available to support the relative likelihood of operator mitigation actions.

The RELAP5 – MELCOR comparison results for this test case (no ECCS, no operator mitigation action) are presented in Figure B-13 through Figure B-18, which plot peak clad temperature, RPV pressure, RPV collapsed liquid level, CVCS break mass flow rate, CVCS break integrated mass flow, and DHRS heat removal, respectively, all as a function of time, out to the start of core damage.

The test case results show good agreement between the two codes. RPV pressure, DHRS heat removal, break flow rate, break integrated flow, and RPV liquid level match very well. The PCT begins rising slightly sooner in the RELAP5 simulation and the start of core damage is reached approximately {{   }}[2(a),(b),(c),ECI] ahead of MELCOR. The start of core damage is {{   }}[2(a),(b),(c),ECI] which provides significant time for operator mitigation actions and supports the crediting of containment injection via CFDS in the two sensitivity cases.

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-13. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) – peak cladding temperature

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-14. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) – RPV pressure

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-15. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) –
RPV collapsed liquid level (TAF at 0 feet)

{{

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

Figure B-16. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) –
CVCS break mass flow rate

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-17. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) – CVCS break integrated mass flow

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-18. CVCS Outside Break Test Case (No Credit for ECCS or Operator Mitigation) – DHRS heat removal

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

TR-0915-17772-NP
Rev. 1

B.3   Example Source Term Evaluation

Source term evaluations are performed on the two CVCS outside break sensitivity cases.

Sensitivity Case A – As described in Section B.1, this accident is a break in the CVCS injection line outside containment; both CVCS isolation valves fail to close, CVCS check valve fails to close, both trains of DHRS actuate, no credit for ECCS, and credit for operator action for containment flooding via CFDS at 8 hours.

The results are presented in Figure B-19 through Figure B-25, which plot peak clad temperature, RPV pressure expanded x-scale, RPV pressure expanded y-scale, RPV collapsed liquid level, coolant mass distribution, CNV liquid level, and balance of power, respectively, all as a function of time, out to 72 hours.

As shown in Figure B-19, no core damage occurs out to 72 hours. From Figure B-20 and Figure B-21, it is seen that RPV pressure decreases rapidly in the first 2 to 3 hours during the blowdown phase (liquid flow out the break). Pressure then declines more slowly, reaching ~40 psia at 8 hours, at which point CFDS is actuated. Upon CFDS actuation, the pressure declines more rapidly due to cooling from heat removal out of containment (i.e., through the RPV wall to the CFDS liquid in the containment annulus to the CNV wall and then to the reactor pool surrounding the CNV), reaching a quasi-equilibrium with outside (atmospheric) pressure at just beyond 10 hours. Starting at this point in time, small amounts of air sporadically enter the RPV over the course of days, which gradually reduces the effectiveness of heat transfer through the RPV wall and the primary side of the steam generators. This gradual reduction in heat transfer effectiveness results in periodic episodes of elevated RPV pressure relative to atmospheric pressure, which can be seen in Figure B-21.

From Figure B-22, RPV liquid level follows a pattern early in the accident that is similar to RPV pressure, reaching a level of about 3 feet above the core at ~10 hours. At this point, RPV level is at a near-stable condition with a very slight decline over the remaining 62 hours, reaching a level just above the top of the core at 72 hours. The slight decline over the remaining 62 hours is due to the slightly elevated RPV pressure noted above, which slowly drives inventory out the break. Figure B-23 also shows similar behavior, with RPV mass at a near-equilibrium condition beginning at about 10 hours and continuing with a very slight decline out to 72 hours.

Figure B-24 shows the water level in containment due to CFDS actuation. The level is zero out to eight hours at which point CFDS is actuated (100 gpm conservatively assumed from one CFDS pump although the CFDS has two pumps with identical flow capacity). This flow fills containment at about 15 hours into the event (7 hours of CFDS pump operation).

From Figure B-25, heat removal out the containment starts to increase at just beyond 8 hours when CFDS is actuated, and the combination of DHRS and containment heat removal are close to decay power after about 30 hours.

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

TR-0915-17772-NP
Rev. 1

Overall, Sensitivity Case A results are similar to the CVCS LOCA Injection Line Outside Containment – Base Case discussed in Section B.2.1, a minor difference being that RPV water level is very slightly declining at 72 hours for Sensitivity Case A versus appearing to be in a long-term, stable equilibrium at 72 hours for the base case. The reason for this difference is the higher heat removal rate from the primary system in the base case where ECCS has actuated and natural circulation between the RPV and the containment has been established. While the heat removal is effective in Sensitivity Case A (through the RPV wall to the CFDS liquid in the containment annulus to the CNV wall and to the reactor pool), due to the gradual air ingress and RPV pressure elevation effect noted above, it slowly degrades and is not able to maintain the more stable, long-term equilibrium seen in the base case. Throughout this event and beyond the time of computer code analysis, CD is not expected to occur.

Sensitivity Case B – Since Sensitivity Case A did not result in a source term, a second sensitivity case has been evaluated. Sensitivity Case B is the same as Sensitivity Case A except the operator action for containment flooding via CFDS is credited at 24 hours instead of 8 hours.

{{

}}2(a),(b),(c),ECI

Figure B-19. CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8 Hours) – peak cladding temperature

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-20. CVCDS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8
Hours) – RPV pressure expanded x-scale

{{

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

Figure B-21. CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8
Hours) – RPV pressure expanded y-scale

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-22. CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8 Hours) – RPV level

{{

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

Figure B-23.   CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8 Hours) – coolant mass distribution

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

TR-0915-17772-NP
Rev. 1

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-24. CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8 Hours) – CNV liquid level

{{

$\}\}^{2(a),(b),(c),ECI}$

Figure B-25.   CVCS Outside Break Sensitivity Case A (No Credit for ECCS, Credit CFDS at 8 Hours) – balance of power

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

TR-0915-17772-NP
Rev. 1

The calculation of Sensitivity Case A thermal-hydraulics and Sensitivity Case B thermal-hydraulics, core damage progression, and radionuclide transport out of the module was performed with a preliminary NuScale full module MELCOR 1.8.6 simulation model. Since core damage occurs in this case, a CVCS separate-effects MELCOR 2.1 model was used for Sensitivity Case B to calculate the fission product aerosol deposition in the CVCS piping (similar to what was done in the SOARCA project). Also, an RXB separate-effects MELCOR 1.8.6 model was used to calculate the accident mitigation effects of the RXB including RXB spray and realistic building exchange rate with the environment. The MELCOR 1.8.6 full module simulations were performed with executable version 3893, MELCOR 2.1 simulations with executable version 5392, and MELCOR 1.8.6 RXB simulations with executable version 3964. The methodology for these separate-effects models is discussed in Section 4.1.1 of the main report.

Thermal-hydraulic results for Sensitivity Case B are given in Figure B-26 through Figure B-33, which plot peak clad temperature, RPV pressure, RPV pressure expanded x-axis, RPV pressure expanded y-axis, RPV-collapsed liquid level, coolant mass distribution, CNV liquid level, and balance of power, respectively, all as a function of time.

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-26.   CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – peak cladding temperature

{{

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

Figure B-27. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – RPV pressure

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-28. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24
Hours) – RPV pressure, expand X-axis

{{

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

Figure B-29. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24
Hours) – RPV pressure, expand Y-axis

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-30. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – RPV level

{{

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

Figure B-31.CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – coolant mass distribution

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-32. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – CNV liquid level

{{

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

Figure B-33. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – balance of power

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

TR-0915-17772-NP
Rev. 1

As is evident from Figure B-26, core damage starts at about {{ }}[2(a),(b),(c),ECI] in Sensitivity Case B. From Figure B-27, Figure B-28, and Figure B-29, it is seen that the RPV pressure decrease is slower out to 24 hours compared to Sensitivity Case A (where CFDS was actuated at 8 hours) which results in additional coolant inventory loss out the break compared to Sensitivity Case A. Figure B-30 shows that RPV level has decreased to ~3 feet below TAF at 24 hours. It should be noted that the time between core uncovery (24 hours) and the time to CD ({{ }}[2(a),(b),(c),ECI]) is significant ({{ }}[2(a),(b),(c),ECI]).

Due to CFDS actuation at 24 hours, RPV pressure tends to level off at near outside (atmospheric) pressure and RPV coolant mass (Figure B-30) has a similar behavior. Figure B-32 shows water level in containment due to CFDS actuation. From 24 to 30 hours, the RPV coolant mass is essentially stable. However, by 30 hours, due to the air ingression effect and resulting slight reduction in effectiveness of heat transfer through the RPV wall and primary side of the steam generators, there is an increase in RPV pressure relative to atmospheric pressure as can be seen on Figure B-29. This elevated pressure drives additional inventory out the break. Given that the core was already uncovered (~3 feet below TAF at 24 hours), this additional inventory loss causes the upper regions of the fuel to begin heating up and generating hydrogen. The hydrogen generation increases system pressure, which again drives flow through the break, increases uncovery of the core, and leads to core damage starting at about {{ }}[2(a),(b),(c),ECI].

The radionuclide release results for Sensitivity Case B are given in Figure B-34 to Figure B-39. These figures show radionuclide release from the fuel, radionuclide release to the RXB, fission product release distribution in various locations for xenon, cesium, and iodine, and radionuclide release to the environment, respectively, all as a function of time. Table B-1 shows the MELCOR model that was used for each of the plots on Figure B-34 to Figure B-39.

Table B-1.   MELCOR models used for Figure B-34 to Figure B-39 plots

Parameter	Figure Numbers	MELCOR Model Used
Release from Fuel	B-34, B-36, B-37, B-38	Full module
Release to RXB for xenon	B-35	Full module
Release to RXB (everything but xenon)	B-35	CVCS separate-effects
In RPV	B-36, B-37, B-38	Full module
In CVCS	B-37, B-38	CVCS separate-effects
In RXB	B-36, B-37, B-38	CVCS separate-effects
In Environment	B-36, B-37, B-38, B-39	RXB separate-effects

From Figure B-34, fission product release from the fuel starts just prior to 40 hours. Volatile releases from the fuel (xenon, cesium, tellurium, iodine, and to a lesser extent molybdenum) are high as is typically predicted by MELCOR full core damage events.

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

TR-0915-17772-NP
Rev. 1

Figure B-35 shows the radionuclide release to the RXB. The xenon release to the RXB is high, with the other radionuclide group releases being significantly lower than what was released from the fuel due to retention in the RPV and CVCS line.

Figure B-36 shows the xenon distribution. As expected, nearly all of the xenon that is released from the fuel is transported to the RXB and the environment.

Figure B-37 and Figure B-38 show the cesium and iodine distributions, respectively, in the RPV, CVCS, RXB, and the environment. Between 20 and 25 percent of the cesium is retained in the RPV and over 75 percent is retained in the CVCS piping. A small fraction of the cesium is released into the RXB, approximately 1 percent as seen in Figure B-35 and Figure B-38. About 30 percent of the iodine is retained in the RPV and close to 70 percent is retained in the CVCS piping. A small fraction of the iodine is released into the RXB; under 2 percent as shown in Figure B-35 and Figure B-37.

A similar release occurs for tellurium. There is significant surface area (high surface to volume ratio) in the flow path (control rod drive mechanisms above the core and in the riser, steam generator tubes) for fission product aerosols that do not enter the broken CVCS line which promotes aerosol deposition in the RPV. The separate-effects CVCS MELCOR model also predicts significant aerosol deposition in the CVCS line due to turbulent deposition and impaction. The liquid level in the containment annulus from CFDS actuation at 24 hours increases quickly as shown in Figure B-32, rising to the elevation of the CVCS penetration through the RPV wall (~15 feet above the core) in about 2 hours, and essentially covering the entire CVCS line in about 6 hours (~30 hours after accident initiation). Thus, the CVCS line is submerged during the release, ensuring minimal revaporization of the deposited volatiles. It should also be noted that significant radionuclide retention in the CVCS line is commensurate with its small two-inch diameter size.

Figure B-36 through Figure B-39 show the radionuclide release to the environment, which is based on the RXB separate-effects MELCOR model. Aerosol removal occurs due to the RXB spray system. There is also some fission product holdup when realistic RXB air exchange rate with the environment is modeled. The iodine release to the environment, for example, is ~0.1 percent (a reduction by a factor of approximately 16 from just under 2 percent).

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-34. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – radionuclide release from fuel

{{

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

Figure B-35. CVCS outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – radionuclide release fraction in RXB

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

TR-0915-17772-NP
Rev. 1

{{

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

Figure B-36. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – xenon distribution

{{

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

Figure B-37. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours) – cesium distribution

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

TR-0915-17772-NP
Rev. 1

{{

}}[2(a),(b),(c),ECI]

Figure B-38.   CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24
Hours) – iodine distribution

{{

}}[2(a),(b),(c),ECI]

Figure B-39.   CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24
Hours) – radionuclide release to environment

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

TR-0915-17772-NP
Rev. 1

Overall, Sensitivity Case B shows that the source term is slow and small, with core damage starting at {{   }}$^{2(a),(b),(c),ECI}$ and iodine release to the environment of about 0.1 percent of core inventory.

B.4   Example Dose Evaluation

A dose evaluation was performed for the source term from Sensitivity Case B using MACCS version 3.10.0 and the methodology from main report, Section 4.2. This is a break in the CVCS injection line outside containment, both CVCS isolation valves fail to close, CVCS check valve fails to close, both trains of DHRS actuate, no credit for ECCS, credit operator action for containment flooding via CFDS at 24 hours. Analysis is performed with MELCOR models that account for CVCS injection line separate-effects aerosol deposition as well as RXB spray removal and realistic RXB air exchange rate with the environment. As with all the analyses presented in Appendix B, the calculated doses are presented solely to illustrate the EPZ size methodology and the robustness of the NuScale design. However, this example of determining dose from a single sequence does not represent the full methodology. The primary method to evaluate dose is to calculate the probability of dose exceedance using all of the more severe sequences. Since dose consequence evaluation is only done for one accident source term in this appendix, a comparison of the maximum distance to 200 rem over all weather trials is performed, which is a simple and conservative representation of the probability of dose exceedance. These numerical values are not intended to form the basis for any NuScale design-specific plume exposure EPZ size.

The dose evaluation result is shown in Figure B-40. Key aspects of the evaluation are as follows:

- As shown in Section B.3, Sensitivity Case B reaches core damage at about {{   }}$^{2(a),(b),(c),ECI}$ and results in approximately 0.1 percent of the iodine core inventory being released to the environment.

- The "plume shift without rotation" model option was used in MACCS.

- The dose calculation was based on conservative assumptions of zero plume energy, ground level release, and peak dose on the spatial grid, a surrogate for peak centerline dose. A dose-in-place model was used (i.e., neither evacuation nor relocation outside the EPZ were credited in the exposure phase).

- Shielding and protection factors used were {{   }}$^{2(a),(c)}$

- The DCF file was modified from the original DCF file included with the release of Version 3.10.0 (FGR13 DCF). This file contains factors from FGR-11 for inhaled exposure, FGR-12 for external exposure, and FGR-13 for cancer risks. The modification implemented SOARCA best practices with respect to residual cancer risk. However, cancer risk is not considered in this analysis and the DCFs for red bone marrow are unchanged.

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

TR-0915-17772-NP
Rev. 1

- The dose at 0.3 miles is approximately {{ }}$^{2(a),(b),(c),ECI}$ whole body acute dose[6] for the Peach Bottom site which has the limiting meteorology as compared to the Surry site.[7] The dose at the estimated site boundary for a NuScale plant is approximately {{ }}$^{2(a),(b),(c),ECI}$ to the whole body. There is significant margin to the 200 rem whole body acute dose metric for Criterion c.

{{

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

Figure B-40. CVCS Outside Break Sensitivity Case B (No Credit for ECCS, Credit CFDS at 24 Hours – acute whole body dose versus distance using Peach Bottom meteorology)

---

[6] The acute whole body dose is approximated as the red bone marrow dose using NUREG-0396 and is shown for illustrative purpose only.

[7] Doses inside the 0.3 mile distance may need to be adjusted to address building wake and other effects as noted in the main body, Section 4.2.1. Furthermore, the dose at all distances would be calculated for an actual COL application site when a site-specific EPZ size is determined using this methodology.

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

TR-0915-17772-NP
Rev. 1

Appendix C.   Example Application of EPZ Methodologies

C.1   Example Sequence Screening

This section presents an example of determining the necessary accident sequences to be evaluated for the NuScale EPZ size basis, applying the methodology from Section 3.4. The results, as a representative analysis, are based on PRA results from Rev. 0 of the NuScale FSAR (Reference 6.4.17) which is part of the NuScale DCA submitted to the NRC in January 2017. The examples are for illustration only and NuScale is not seeking NRC approval of the information in this appendix. This example uses input from the following PRA hazard models: internal events, internal flooding, internal fire, high winds, external floods, and low power and shutdown. All CDF values presented are point-estimates, following the methodology of Section 3.4.1.

There are a total of 225 individual accident sequences that result in core damage which are above a truncation frequency of 1E-15 per module year. These sequences result in a total point estimate CDF of 9.1E-8 per module year. {{

}}[2(a),(c)]

Table C-1 contains the 13 highest CDF sequences (1-13); along with two other sequences (14 and 15) selected to show that some accidents with high importance for operating reactors are screened out for the NuScale design. Note that the sequence numbering is arbitrary and only used for the purposes of this appendix. {{

}}[2(a),(c)]

Section C.2 contains an example assessment of defense-in-depth. {{

}}[2(a),(c)]

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

TR-0915-17772-NP
Rev. 1

{{

}}²(a),(c)

Table C-1.   PRA sequences with CDF contributions

{{

}}²(a),(c)

C.2   Example of Defense-in-Depth Assessment of Accident Sequence

Following the methodology of Section 3.5, an example defense-in-depth assessment is performed {{

}}²(a),(c)

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

TR-0915-17772-NP
Rev. 1

The attributes descriptions from Section 3.5 are not repeated; instead, only the technical information needed to evaluate each attribute and evaluation itself is provided.

C.2.1   Criterion 1

{{

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

C.2.2   Criterion 2

{{

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

C.2.3   Criterion 3

{{

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

C.2.4   Criterion 4

{{

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

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

TR-0915-17772-NP
Rev. 1

{{

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

C.2.5   Criterion 5

{{

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

C.2.6   Overall Defense-in-Depth Assessment

{{

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

Table C-2.   Defense-in-depth overall assessment

{{

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

C.3   Example Application of Uncertainty Analysis Method

C.3.1   Selection of Accident Sequence

{{

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

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

TR-0915-17772-NP
Rev. 1

C.3.2   Selection of Uncertainty Parameters

The source term and dose evaluations to be performed to support the NuScale plume exposure EPZ sizing analysis are similar to the SOARCA analyses. In both analyses, the MELCOR code is used to evaluate source terms for given accident sequences and the MACCS code is used to evaluate doses.

The similarity between the two analyses makes the uncertainty parameter list for the SOARCA analyses a reasonable starting point for the NuScale EPZ sizing analysis. The following paragraphs describe an example starting from the list of SOARCA parameters that considered NuScale-specific design features to create an example list of uncertainty parameters. A user of this methodology should follow the same process, after screening of accident sequences, to determine a final list of parameters. In SOARCA, the parameters were sorted into the following categories to ensure all phases of the accident were included:

For MELCOR:

- sequence of the event
- in-vessel accident progression
- ex-vessel accident progression
- containment behavior
- fission product release, transport, and deposition

For MACCS:

- fission product aerosol deposition
- shielding factors
- dispersion parameters
- weather trials (random)

First, the uncertainty parameters selected for SOARCA uncertainty analyses are screened to eliminate those that are not relevant to the NuScale design or to the EPZ uncertainty analysis methodology. Thus, some SOARCA parameters are removed. An example of parameters removed because they are not relevant to the NuScale design are those associated with the railway door, or the drywell, from the MELCOR list. An example of parameters removed because they do not apply to the NuScale methodology are those associated with health risks or mitigative actions from the MACCS list.

Second, the remaining uncertainty parameters are examined to further eliminate the ones that are not expected to have significant impact on source terms for NuScale design. {{

}}[2(a),(c)]

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

TR-0915-17772-NP
Rev. 1

Hydrogen-related parameters for containment behavior in the SOARCA list are moved into the RXB behavior category since for the NuScale design, containment hydrogen combustion is not significant due to extremely limited oxygen concentration, as the containment is maintained at negative pressure conditions during normal operation.

The SOARCA uncertainty parameters that survived the two screening processes are now retained for the NuScale EPZ sizing uncertainty analysis. Next, additional parameters are identified in relation to the candidates for accident sequence selection for the NuScale uncertainty analysis. {{

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

For MACCS, beyond site- and sequence-specific parameters such as weather and release height, there shall be consideration of parameters that impact near field effects, such as building wake effects. As discussed in Section 4.2.4, MACCS will be compared against the result of an NRC-qualified code within 500 meters of the release point and thus additional parameters shall be added to the uncertainty analysis. The parameters selected should be those used in the MACCS comparison or those used to modify MACCS input if such change is necessary.

Examples of uncertainty parameters to be considered for the NuScale EPZ sizing uncertainty analysis are summarized in Table C-3. It is expected that with further refinement more of the parameters included here would be removed as not expected to have significant impact on the selected accident sequences. Parameter distributions are not included.

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

TR-0915-17772-NP
Rev. 1

Table C-3.   Examples of parameters considered for EPZ sizing uncertainty analysis

MELCOR	MACCS
Epistemic Uncertainty	**Epistemic Uncertainty**
Sequence Issues	**Deposition**
Battery duration	Wet deposition model
Total decay heat	Dry deposition velocities
In-Vessel Accident Progression	**Shielding Factors**
Zircaloy melt breakout temperature	Groundshine
Molten clad drainage rate	**Latent Health Effects**
Fuel failure criterion	Inhalation dose coefficients
Radial molten debris relocation time constant	**Dispersion Parameters**
Radial solid debris relocation time constant	Crosswind dispersion coefficients
Material properties: eutectic temperatures for zircaloy oxide and uranium oxide	Vertical dispersion coefficients
Containment and Steam Generator Behavior	Plume release height
Condensation – effect of noncondensable gas on condensation rate	Initial dispersion coefficients
Failure location and size	RG 1.145 model inputs for plume meander
Containment leak rate	Plume meander factors
Reactor Building Behavior	**Aleatory uncertainty**
Reactor building leakage rate	Weather trials
Hydrogen ignition criteria	
RXB spray start time and flow rate	
Leakage to bulk regions	
Pool scrubbing DF	
Filter DF	
Chemical Forms of Iodine and Cesium	
Iodine and cesium fraction	
Aerosol Deposition and Transport	
Particle shape factor	
Particle density	

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

TR-0915-17772-NP
Rev. 1

Appendix D.   Example Assessment of Multi-Module Effects

This appendix provides details for the type of accident sequences to be addressed for multi-module effects. It serves as an example of the information that would support multi-module screening as presented in Section 3.7.

D.1   Hazard and Initiating Event Assessment

The evaluation of multi-module accidents requires that all hazards and their associated initiators be assessed against the criteria described in Section 3.7.1. These hazards include the following:

- internal events

- internal fires

- internal floods

- high winds

- other external events

- seismic events

Note: Each hazard described in this section includes both full-power and low power and shutdown.

These hazards are discussed in the following subsections.

D.1.1   Internal Events

The multi-module implications of each initiator are assessed in Table D-1 below.

Table D-1.   Multi-module implications of internal initiators

Initiator	Multi-Module Implications
CVCS LOCA Inside Containment—Charging Line CVCS LOCA Outside Containment—Charging Line CVCS LOCA Outside Containment—Letdown Line	CVCS is module specific and there is no coupling between safe shutdown functions in multiple modules.
Spurious Opening of an ECCS Valve	ECCS valves are located on RPVs in separate modules and there is no coupling between safe shutdown functions in separate modules.
Loss of DC Power	Common-cause initiator (two buses required to fail), no design-specific coupling mechanism between safe shutdown functions in separate modules.
LOOP	Site-wide initiator, independent safe shutdown functions in multiple modules are not compromised (see Figure 3-4).

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

TR-0915-17772-NP
Rev. 1

Initiator	Multi-Module Implications
Steam Generator Tube Failure	Single-module initiators, initiator does not directly compromise safe shutdown functions in multiple modules
LOCA Inside Containment	
Secondary Side Line Break	
General Reactor Trip	Potential site-wide initiator, independent safe shutdown functions in multiple modules are not compromised
Loss of Support Systems	Potential site-wide initiator, fail-safe safe shutdown functions operate independently.

D.1.2   Low Power and Shutdown

Low power and shutdown configurations present risks for multi-module accidents because of the module transport and refueling operations. While only one module can be transported or refueled at a time, it passes in physical proximity to other modules during the refueling process and is detached from its normal supports.

A failure of the reactor building crane (RBC) during transport has the potential to result in a dropped module. It should be noted that the module is only transported under the following conditions:

- CNV flooded up to the pressurizer baffle plate

- ECCS valves opened and containment isolated

- All control rods inserted into the core, which is subcritical

As such, the release potential of a dropped module under these conditions is diminished compared to an operating module.

NuScale module refueling operations are expected to be performed on a regular basis (one module every two months). Only one module can be transported or refueled at any one time. In addition, module decay heat removal during shutdown does not rely on a separate powered system.

D.1.3   Internal Fires

Internal fires are events that initiate within the plant boundary and can propagate to one or more compartments. Internal fires can include wide-ranging effects on multi-module initiators as well as shared systems between modules. Table D-2 assesses the multi-module implications of internal fire-induced initiators.

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

TR-0915-17772-NP
Rev. 1

Table D-2.   Multi-module implications of induced fire events

Category	Comments	Multi-Module Implications
Fire-induced transient	Base case for internal fire events. Transients may include or exclude the availability of support systems.	Shutdown signal to all affected modules. Both DHRS and ECCS are nominally available to provide cooling in all modules.
Fire-induced LOOP	A fire that results in a loss of the normal AC power supplies to the plant.	Shutdown signal to all affected modules. DHRS and ECCS available.
Fire-induced ECCS demand	Extension of the transient case where a fire-induced failure that also actuates the ECCS valves. For modeling purposes, this event also includes induced loss of DC power.	Possible multi-module implications if fire affects compartments with system (e.g., control systems) cabling for multiple modules. Once the passive ECCS valve setpoint is reached through depressurization, a fire-induced failure that prevents passive actuation is considered incredible since the same fire event has caused the opposite configuration.
Fire-induced LOCA inside containment	This case is an extension of the transient case where there is a spurious operation signal sent to the CVCS makeup pumps resulting in the potential to over-pressurize the RPV, demanding RSV cycling.	CVCS is module-specific; however, pumps associated with different modules are located in the same fire compartment. Mitigated by successful CVCS isolation or ECCS actuation.

Fire compartments are designed (with the exception of the control room, the area under the bioshield, and the reactor module) to ensure separation between Division I and II equipment, including engineered safety features actuation system, so that a single-compartment fire cannot affect both trains of the affected system. However, some postulated fire sequences result in both divisions being affected, particularly multi-compartment fire sequences. Multi-compartment fires involve fire spreading and the failure of compartment barriers, and thus have lower frequencies.

Internal fires have the potential to induce initiators (such as a LOOP) on more than one module. The only shared system susceptibility to a single internal fire event is through the backup power supply system and nonsafety-related RPV and CNV makeup systems. The control circuits for each division of these systems are independent, meaning that both would need to experience a separate hot short (spurious signal failure) to fail the system.

Other fire initiators result in a spurious ECCS actuation signal. ECCS cabling for all 12 modules is expected to run through the same compartments. Following shutdown and RPV depressurization, fail-safe positions of all ECCS valves cannot be guaranteed since hardware failures of the valve itself could prevent some valves from opening. Spurious opening would be demanded instantaneously, at which point the solenoid de-energizes. Opening of the ECCS valve would only occur once the differential pressure (between the RPV and the CNV) decreases below the inadvertent actuation block (IAB) setpoint. Conversely, fire-induced failure to open would require a hot short occurring at the moment of the initiator, and persist until the IAB setpoint is reached for core cooling to be affected by the incomplete actuation.

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

TR-0915-17772-NP
Rev. 1

This would require a prolonged failure to clear the hot short for any affected module. This fault clearing function is module-specific and no event coupling mechanisms exist that would cause an elevated likelihood of fault clearing failure to occur in more than one module.

D.1.4   Internal Floods

Similar to internal fire risk, internal flood risk is modeled using induced initiators and mapping equipment in affected compartments.

In accordance with the approach outlined in Section 3.7.1.4, a flooding sequence only presents a multi-module interaction risk if the following criteria are met:

- the flood-induced initiator affects multiple modules simultaneously

- there is a NuScale design-specific vulnerability to mitigating function impairment for more than one module

Electrical protections are assumed to ensure that SSCs transition to fail-safe positions. NuScale's passive safety features and no reliance on electrical power for safety-related systems mean that there is no additional coupling mechanism from internal flooding beyond correlations between random failures.

D.1.5   High Winds

High winds, including tornadoes and hurricanes, have limited potential to affect NuScale SSCs beyond an induced LOOP. While the induced LOOP would affect all modules simultaneously, such a condition would also occur at a multi-unit site. Consequences for multi-unit initiators are not combined.

Because the NuScale RXB is a Seismic Category I and aircraft impact resistant structure, it is not susceptible to damage from high winds, wind-generated missiles, or damage from other buildings. RXB structural damage is, therefore, screened out for high wind hazards along with any potential multi-module implications.

High winds do have the potential to damage off-site power equipment, leading to a LOOP for all 12 modules. Both the alternate AC power system and BDGs are susceptible to failure from high winds, which implies that extreme high wind events could lead to a prolonged loss of AC power.

However, safety systems will assume their fail-safe position on loss of AC power (and subsequent loss of DC power on battery depletion) and, therefore, a high wind event introduces no additional coupling mechanism between the random failures in separate modules that would need to occur in a core damage sequence.

D.1.6   Other External Events

The NuScale design includes several features that preclude adverse conditions from external hazards, and these features are common to all modules. They include:

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

TR-0915-17772-NP
Rev. 1

- an RXB with reinforced exterior walls designed for missile protection

- a reactor pool that is located below grade and lined with stainless steel

- no reliance on external power to provide core and SFP cooling

These features preclude multi-module propagation of accident sequences following an external event. Similar to high wind events, the direct effects of these other external events, in general, are limited losses of off-site power.

D.1.7   Seismic Events

In accordance with the approach outlined in Section 3.7.1.7 seismic events present a unique challenge because of their site-wide effects and hazard-specific failure modes. RXB structural failures, in particular, have the potential to affect multiple modules. Similarly, seismic failures of identical components located in different reactor modules can be correlated.

Potential multi-module implications of structural failures caused by a seismic event are listed in Table D-3. Pool drain is not included as NuScale has determined that it is not a credible event.

Seismic initiators are categorized by the PGA value. The occurrence frequencies associated with PGAs are site-dependent and are not quantified in this methodology. Seismic initiators are first evaluated for structural effects, followed by other induced initiators.

Chapter 19, Section 19.1.5.1 of NuScale's FSAR (Reference 6.4.17) documents the plant-level fragility and HCLPF using the MIN-MAX method for seismic cutsets. These results show that the RBC has the lowest HCLPF value. Since a crane failure is assumed to lead directly to core damage, this event is the controlling failure mode.

Table D-3.   Multi-module implications of seismic structural failures

Structural Event	Controlling Failure Mode	Assumed Consequence	Multi-Module Implications
RXB crane	Bridge seismic restraint weldment yielding	Core damage/large release	One module under the crane hoisting mechanism affected
RXB walls	Out-of-plane shear cracking at base of outer E-W wall		All 12 modules potentially affected
Reactor module support failure	Shear failure of multiple shear lugs	Core damage/large release	Module-specific, potential correlated failures in other modules
Reactor bay wall	Flexural failure	Core damage/large release	Two modules (one on either side of bay wall) affected
Bioshield – horizontal shear flexure – normal operation	Horizontal shield slab bending failure	Core damage/large release	

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

TR-0915-17772-NP
Rev. 1

Structural Event	Controlling Failure Mode	Assumed Consequence	Multi-Module Implications
Bioshield – pool wall bolt failure – normal operation	Shear failure of pool wall anchor bolts	Core damage/large release	Module-specific
Bio shield – horizontal shear flexure[1]	Bending failure of both stacked shield slabs	Core damage/large release	
Bio shield – pool wall bolt failure[1]	Shear failure of pool wall anchor bolts	Core damage/large release	

1. This event is only possible during the infrequent configuration when a second bioshield is stacked on the existing bioshield for refueling of an adjacent module.

D.2 Consequential Initiating Events

This section describes the potential for initiators occurring in one module to induce a separate initiating event in another which is denoted as a consequential effect.

D.2.1 High Energy Line Break Consequential Effects

High energy line breaks (HELBs) are included in the module-specific secondary line break event that can cause a harsh environment in the RXB and cause a transient initiator in any number of the other modules. While SSCs necessary for safe shutdown are protected against pipe whips and fluid jets, the energy release from the HELB can result in high pressures and temperatures inside the RXB. This would result in an administrative shutdown, which may affect multiple modules. The robustness of the Seismic Category 1 design of the building and the absence of design-specific coupling mechanisms following the consequential transients permits these sequences to be screened from analysis.

D.2.2 Fire Consequential Effects

The effects of fires are addressed in Section D.1.3. No separate consequential effects from fire initiators are assessed.

D.2.3 Flood Consequential Effects

Flood events affecting one module have the potential to induce transients in other modules, leading to multiple demands to shut down. Beyond that, the fail-safe design of the DHRS, ECCS, and containment isolation valves results in no flood-specific multi-module dependencies.

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

TR-0915-17772-NP
Rev. 1

D.2.4   Seismic Consequential Effects

Seismic events are wide-ranging by nature and have the potential to induce other initiating events, such as fires and floods. While it is possible for seismic events to affect one module and the consequential fire or flood to affect separate modules, the seismic event does not introduce a new coupling mechanism for the failures discussed in Section D.1.7.

D.3   System Failure Mode Analysis

All accidents involving physical interactions between modules originate from individual SSC failure modes. This section provides an overview of system failure modes and multi-module screening criteria.

D.3.1   Random and Nonseismic Failures

D.3.1.1 Mapping of SSCs with Inter-Module Interaction Potential

NPMs are normally located in a common reactor pool, in two rows of six modules. Each NPM in a row is separated by a bay wall, which is attached at the floor and the reactor pool wall. The reactor pool layout is shown in Figure D-1 below.

{{

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

Figure D-1. Inter-module dimensions in reactor pool

The rows of opposing modules as well as the module transit path are a design-specific NuScale feature.

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

TR-0915-17772-NP
Rev. 1

D.4   Dropped Module Interaction Example

This section describes an example of a multi-module accident involving crane failure, leading to a dropped module, which then topples and strikes two other modules. While the module that is dropped during transport is shut down and the CNV is partially flooded, passive cooling of a horizontal module may be insufficient to prevent core damage.

The sequence originates from an uncontrolled drop of a module in transit held by the crane. The module becomes completely severed from the crane and falls to the floor of the reactor pool, damaging its support skirt and causing it to topple in a random direction. It should be noted that the module is lifted less than 3 feet throughout its movement in the pathway between operating modules until it is ready to be lifted onto the containment flange tool. At that location, the crane lifts the module 30 feet. An array of possible configurations and collision possibilities is shown in Figure D-2.
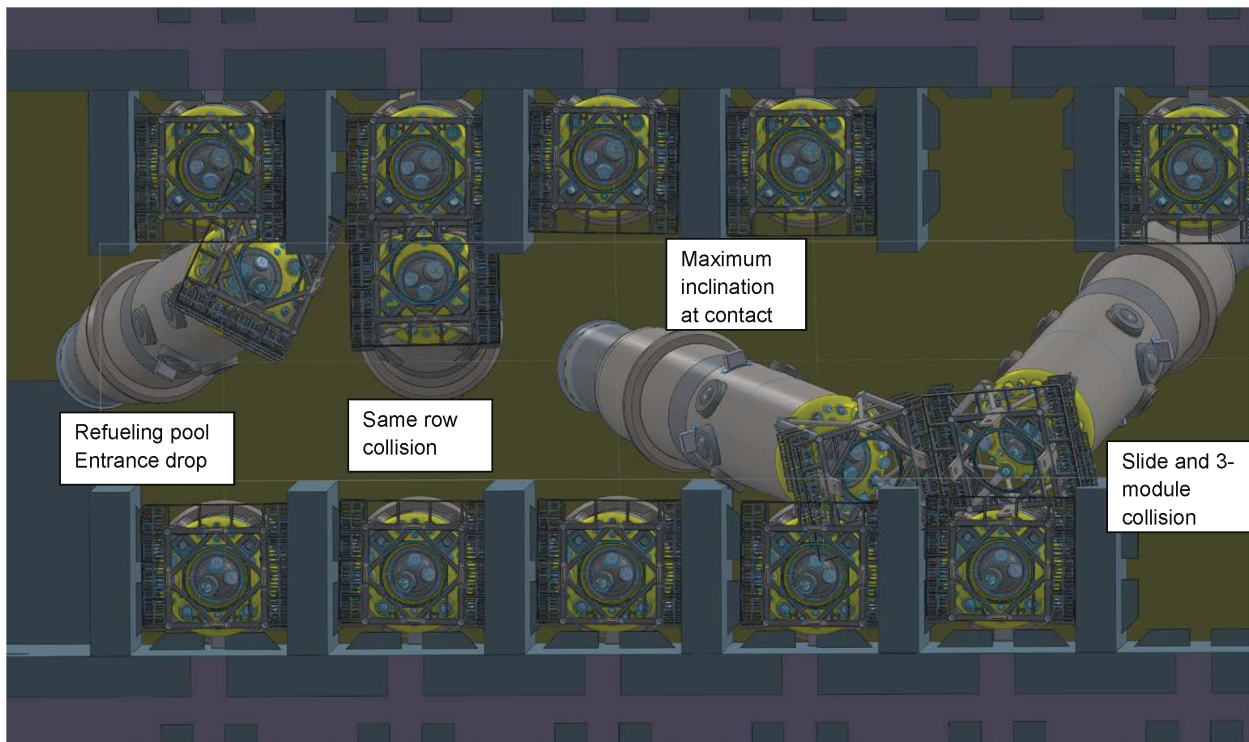

Figure D-2. Dropped module interaction geometry

The height of the NPMs and physical constraints of the bay walls results in the initial collision between a toppling module and operating module occurring at the module platform level. The impact velocity of the toppling module is highly dependent on the distance travelled by the top of the module. Beyond a critical impact velocity, the resulting stress is sufficient to exceed the material yield strength of the operating module platform structure, and cause damage to the piping located at the platform. This has the potential to induce a LOCA outside containment initiating event in the operating module.

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

TR-0915-17772-NP
Rev. 1

Following the collision with an operating module, the dropped module may reach an inclination sufficient to cause it to slide backwards along the floor and strike a third module, as shown in the rightmost illustration of Figure D-2. Sliding after collision may occur if the toppled module is at sufficient incline for the resulting forces to overcome the friction of the support skirt against the floor and the platform against the wall or other module. The sliding module may collide with the base of a third module, which could damage the support skirt. Consequently, the impact could induce additional failures in the third module.

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

TR-0915-17772-NP
Rev. 1

Appendix E. Design-Specific Methodology for Operationally-Focused Mitigation Capability

E.1 Introduction

The NuScale design-specific, operationally-focused mitigation capability provides additional accident mitigation capability for EPZ that is based on deterministic rather than probabilistic considerations. Credit for additional mitigation is optional for more severe accidents. This appendix describes the purpose of the mitigation capability in the context of the EPZ basis, provides an overview of the capability, and presents an example of a methodology that an applicant could employ to confirm the effectiveness of the capability in supporting the EPZ sizing basis.

Purpose

When credited, the purpose of the NuScale design-specific operationally-focused mitigation capability in the context of the EPZ basis is to do the following:

1. Support the defense-in-depth aspect of the risk-informed EPZ methodology, by highlighting the deterministic aspects of the operationally-focused mitigation capability

2. Take advantage of the relatively slow accident progression in the event of a severe accident and the fact that the operationally-focused mitigation capability is being developed in an integrated fashion in parallel with the design.

3. Support the consequence orientation (Assumption 3 in Section 3.1) of the risk-informed EPZ methodology (i.e., important aspects of the capability are intended to apply once core damage has occurred)

Plant design

The NuScale plant is an innovative design based on 50 years of practical application of light-water-cooled pressurized-water reactor technology. The design incorporates several features that reduce complexity, improve safety, and enhance operability. The NuScale design philosophy includes:

• the incorporation of proven standard technology

• smaller reactor core size

• a below-grade containment immersed in an UHS pool of water.

• passive safety systems

• no operator action required for at least 72 hours following a DBA

• no reliance on AC or DC power for all design bases accidents

• After 72 hours, reactor pool evaporation, pool water boil-off, and air cooling of containment are capable of providing indefinite reactor module decay heat removal without operator action, AC or DC power, or makeup water.

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

TR-0915-17772-NP
Rev. 1

- several months supply of water in the UHS pools before spent fuel can be uncovered from boiling off inventory

Design-basis accident mitigation and core damage prevention

The NuScale design relies on automatically actuated, passive safety systems to mitigate the consequences of accidents. The passive design relies on pressure vessels, valves, piping, and heat exchangers in conjunction with natural convection and conduction to remove decay heat and contain fission products. The design does not require makeup and can continue to remove heat from the module based on the water inventory at the accident initiation for an indefinite period of time. The four primary systems that, by design, mitigate accidents are: decay heat removal; emergency core cooling; ultimate heat sink; and containment. The only components that change state in these systems to initiate the safety function are valves. The valves in these systems only have one safety position and they fail to that position when power is removed or lost. All of the design-basis events are successfully mitigated without operator intervention required resulting in no core damage.

Beyond-design-basis event mitigation

Several BDBEs have been analyzed that include multiple and concurrent failures of these passive systems. These can be categorized by failure modes into three groups: (1) failure of one set of ECCS valves (either all RRVs or all RVVs) to properly actuate, (2) containment bypass events, and (3) complete failure of both the DHRS and the RSVs. In the case of an incomplete ECCS actuation, the water remains in containment and does not return to the vessel because all ECCS vent valves or both ECCS recirculation valves fail to open. In the case of containment bypass, the RCS inventory is lost outside of containment. In the case of a complete failure of the DHRS and the RSVs, all of the heat removal methods are unavailable to the RCS. The failure of the DHRS also prevents depressurization of the RCS and keeps pressure above the ECCS inadvertent actuation block for the ECCS vent and recirculation valves, which prevents them from opening. The failure of the RSVs to open prevents RCS inventory from being discharged to containment, which would couple the RCS to the UHS and eventually would lead to RPV overpressurization.

The first two categories of BDBEs can be mitigated by operator action to restore CVCS makeup capability and provide inventory to the RCS. This is only effective if AC power is available to the CVCS makeup pumps and the system is intact enough to provide a flow path. If this is not available, both result in core damage due to a loss of inventory inside the reactor vessel. The containment bypass event can also be mitigated by adding inventory to the containment with the CFDS.

The third category of BDBEs can be mitigated by adding inventory to the containment with the CFDS, thereby coupling the RCS to the UHS, which will cool down and depressurize the RCS.

In all three of these events core damage takes a relatively long time to occur. Operator mitigation actions can be accomplished within 30 minutes; this was demonstrated during

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

TR-0915-17772-NP
Rev. 1

the NuScale Staffing Plan validation (Control Room Staffing Plan Validation Results, RP-0516-49116 [Reference 6.6.2]).

Subsections E.2 through E.6 provide an example of the steps to be followed by the applicant to demonstrate the effectiveness of the operationally-focused mitigation capability. Although described as requirement, both credit for additional mitigation and use of this methodology are optional. It is expected that much of the work in these steps will have already been completed as part of a COL application and need only be summarized in the EPZ submittal.

E.2   Describe Operator Staffing, Qualification, and Training

The applicant should describe the staffing, qualification, and training of the operators including dealing with beyond-design-basis events. Operators' ability to take appropriate actions will depend largely on the capability of the human system interface to alert them to the degraded conditions, their training, plant equipment on which they will rely, time to perform these actions, and on procedures that direct them on what actions to take.

An applicant will establish minimum staffing and qualification and demonstrate that the design can be safely operated with the minimum staff.

- In the NuScale design, the on-site operations minimum staffing will include licensed and nonlicensed operators as follows:
  - one shift manager (senior reactor operator (SRO) license)
  - one CR supervisor (SRO license)
  - one shift technical advisor (SRO license and technical degree)
  - three reactor operators (reactor operator license)
  - four nonlicensed operators

Training and qualification of operations personnel will be administered using a systematic approach to training. This training will include instruction on the progression of core damaging events, how they are recognized, and actions that can be taken to prevent or mitigate core damage. The training programs would also include the basis of the EPZ sizing.

E.3   Discuss Symptom-Based Procedure and Accident Monitoring Provisions

Applicants should develop procedures to provide consistent and specific direction to the operator for a full range of identified conditions including mitigating actions.

The applicant should:

- identify the symptom-based procedures and guidelines being followed by the operators in the event of degraded conditions.

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

TR-0915-17772-NP
Rev. 1

- describe how these procedures are integrated into a single set for the NuScale design; the colored flow chart can be included as a conceptual example of how this is to be accomplished.

- summarize the accident monitoring provisions in the design and the human system interface to alert the operators to the degraded condition and the need for manual actions.

- discuss how the procedures and monitoring provisions address critical safety functions.

The following is brief description of the types of procedures that will be available to the NuScale operating staff:

Normal operating procedures (NOPs) – Normal operation is defined as plant operation within specified operational limits and conditions. Examples include starting up and shutting down the plant, normal power operation, maintenance, testing, and refueling.

Alarm response procedures (ARPs) – Procedures entered based on receipt of a plant notification alarm or caution. ARPs direct actions to take in response to a particular alarm or caution. The direction taken is generally fairly simple; if a more integrated response is required then the operator is directed to an abnormal operating procedure.

Abnormal operating procedures (AOPs) – Abnormal operations calling for AOPs are off-normal operational states which, because of appropriate design provisions, would most likely not result in the loss of a critical safety function, cause any significant damage, nor lead to accident conditions. Accident conditions are defined as deviations from normal operation more severe than anticipated operational occurrences (AOOs), including DBAs, BDBEs and severe accidents. In abnormal operation the plant is in a situation that represents a potential threat to the integrity of the reactor core but which can be handled by the normal control systems if there are no additional failures.

The NuScale power plant operators will use symptom-based guidelines to identify threats to plant safety functions and provide actions to mitigate threats. These guidelines will be fully integrated and encompass SAMGs and FLEX support guidelines. For clarity, the current terminology of this guidance will be referred to in this report.

ARPs, AOPs, EOPs, SAMGs, and FLEX are all symptom-based procedures. These procedures are described here separately by their current industry designations for clarity but will be integrated into a single set of procedures for the NuScale design.

Emergency operating procedures (EOPs) – These procedures will be symptom-based procedures and will monitor critical safety functions used to prevent core damage and direct action to restore these functions if they are lost. These actions include the operator actions assumed in the PRA. They include both critical safety functions and defense-in-depth safety functions. The three critical safety functions (Table E-1) monitored in the EOPs, are core heat removal, containment integrity, and reactivity control.

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

TR-0915-17772-NP
Rev. 1

Safety function monitoring will be initiated any time a reactor trip or safety system actuation has occurred or conditions indicate that one is required. Actions taken to recover a function are primarily the actions modeled in the PRA fault trees, and are initiated based primarily on the post-accident monitoring variables.

Table E-1.   Post-accident monitoring variables

Safety Function	Major Post-Accident Monitoring Parameters
Containment	- Pressure - Level - Containment isolation valve position
Core cooling	- Core exit temperature - RCS pressure - RCS level - ECCS valve position - Containment level
Reactivity control	Nuclear instrumentation

Monitoring will also include defense-in-depth actions that would be implemented to support safety function restoration. An example is electrical power—although not required to maintain the modules in a safe condition—it is required to perform some of the safety function recovery actions. The NuScale design has several AC power sources available, two BDGs, which can power certain important-to-safety loads and an alternate AC power source which can restore power to any AC electrical bus, and is sized to restart a unit. The NuScale plant is also capable of utilizing a single unit to power the site when a LOOP occurs, this mode of operation is called "island mode."
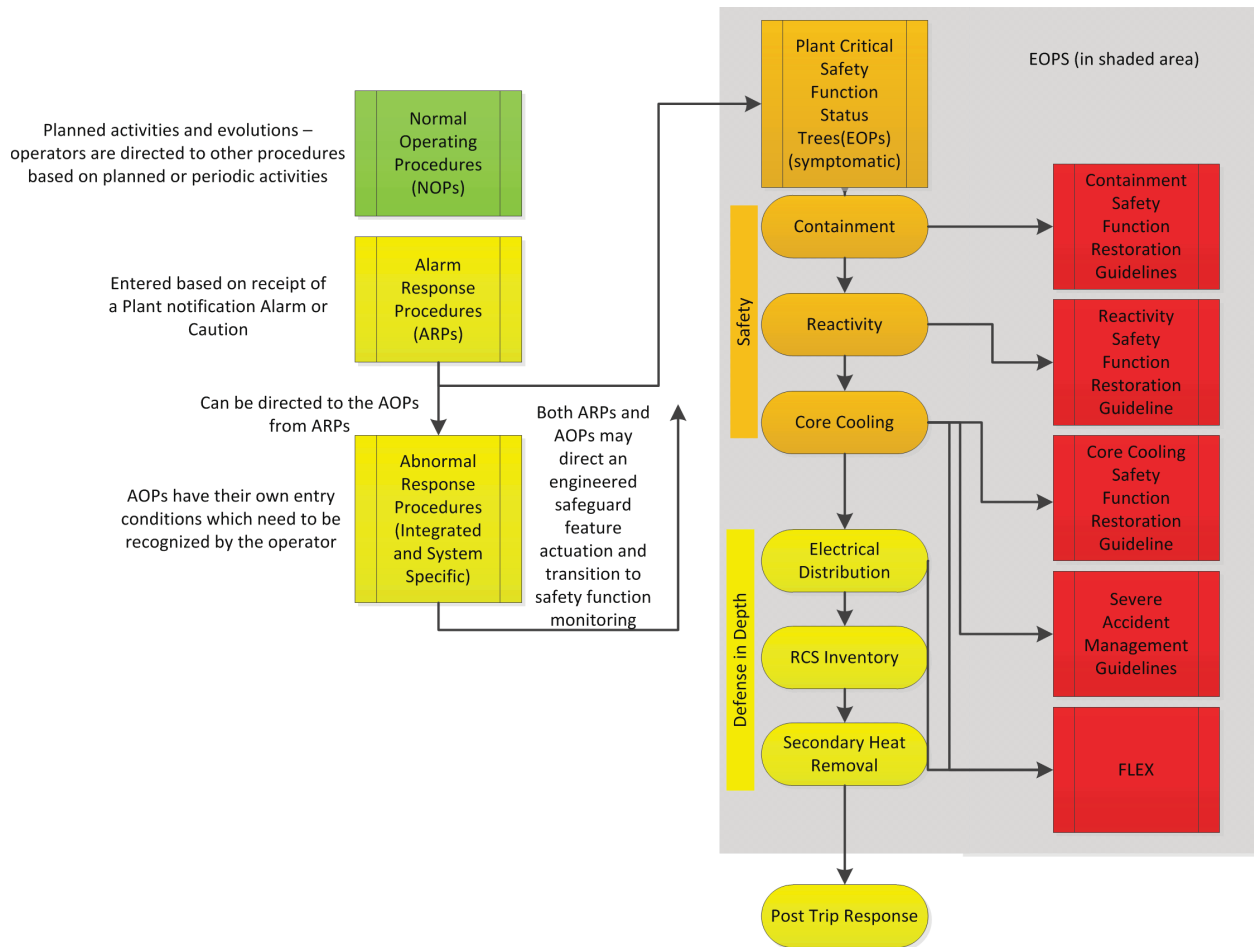
**Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones
at NuScale Small Modular Reactor Plant Sites**

TR-0915-17772-NP
Rev. 1

Figure E-1. Procedure implementation flow diagram

Severe accident management guidelines (SAMGs) – These guidelines provide actions to contain a damaged core, they are not implemented until core damage is imminent or has occurred, these procedures shift the operator's focus from preventing core damage to containing the damaged core. Generally, very high core exit temperatures indicate insufficient water remains in the core and this indication is used to transition to the SAMGs. Given the symptom-based initiation of the SAMGs, NuScale intends to fully integrate these actions into the safety function monitoring trees. An example of a severe accident capability is the use of the RXB spray system for release scrubbing.

FLEX support guidelines – These procedures take actions based on loss of capability; for example, if the plant has lost the ability to makeup to the reactor pool or when it is determined that the site is in an extended loss of AC power (ELAP), the operator takes action to restore the capability. FLEX support guidelines will be developed during the COL application. They will support activities as described in Chapter 20 of the NuScale FSAR (Reference 6.4.17). An example of a FLEX capability is the use of a portable pump to makeup to the SFP.

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

TR-0915-17772-NP
Rev. 1

E.4   Identify Key Recovery Actions and Associated Equipment

The applicant should identify the key recovery actions and associated equipment supporting the various procedures and monitoring provisions including permanently installed equipment, portable on-site equipment, and any off-site and/or regional assets.

In the NuScale design the example recovery actions in Table E-2 were identified.

Table E-2.   Recovery actions

Description	Context
Initiate CFDS injection	Used for LOCA-OC, SGTF, and transients
Initiate CVCS injection	Used for LOCA-IC, LOCA-OC (letdown), transients and secondary steam line break, upon failure of ECCS, and SGTF
Manually isolate containment	Backup action to auto function failure
Locally unisolate and initiate CVCS injection	Local unisolation due to lack of control from a partial loss of DC power
Manually open the ECCS valves	Backup action to auto function failure
Add inventory to the UHS through the SFP assured makeup line	Long-term ELAP action (> 30 days)
Align reactor building spray	Mitigate radiological consequences of a damaged core with a release into the RXB in progress

These were identified by review of the following chapters of NuScale's FSAR (Reference 6.4.17):

1. Human factors engineering (HFE) task analyses results as described in FSAR Chapter 18.4 and its associated reference.

2. The operator actions assumed in the beyond-design-basis PRA as described in FSAR Chapter 19.

3. The operator actions assumed in BDBE evaluation as described in FSAR Chapter 20.

4. Multi-unit design considerations as described in FSAR Chapter 21.

E.5   Demonstrate Effectiveness of Operator Actions

The applicant should perform deterministic, severe accident modeling of accident sequences for which the NuScale design-specific operationally-focused mitigation capability is applied. The purpose of this modeling is to demonstrate the effectiveness of the various operator actions in terminating the accident and/or reducing fission product transport and release in the event of core damage. Examples of this type of modeling are included in SOARCA.

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

TR-0915-17772-NP
Rev. 1

The accident sequences modeled should include sequences for which EOPs have been probabilistically credited to prevent core damage, and sequences screened in for EPZ where SAMGs and FLEX type actions have been considered. This modeling should be supplemented with time studies to show that there is enough time and information for the operator to accomplish the action.

E.6   Address SMR Plant-Specific Operating Experience

The applicant should describe steps that have been taken in design and operations to compensate for the lack of SMR plant-specific operating experience in the context of the EPZ basis. These steps include the following:

- design features that increase the reliability and effectiveness of operator actions under degraded conditions. Example features can be cited such as: smaller, slower source terms; design which minimizes multi-module effects; minimal impact of shared systems on risk; and no need for operator action to prevent or mitigate design-basis events.

- application of industry operating experience in development of the NuScale HFE program that supports designing and inspecting the main control room operator interfaces.

- use of previous license holders in developing NuScale control system design, procedures, conduct of operations, and emergency planning elements.

- use of a state-of-the-art simulator together with an optimized human-system interface (HSI) as input to design and procedure development, training of the operators, and emergency plan development.

- application of previously developed and endorsed emergency action level (EAL) schemes for classification of emergency events and adapting these schemes for SMRs and the NuScale design.

In the NuScale design the following actions were taken to address the lack of plant specific operating experience as it relates to the development of the EPZ.

The smaller source term and longer times to core damage give NuScale operators an advantage over those at existing facilities to take additional time and leverage additional resources to respond to accident conditions. The systems providing direct support to the fission product boundaries and core protection systems are not shared. Internal events are not expected to affect multiple modules. Nonsafety-related systems that are needed for electrical production and the UHS may affect multiple modules. These shared systems for electrical production do not contribute to core damage and the loss of the ultimate head sink has been determined not to be credible. No operator action is credited in any of the design-basis analysis results to prevent or mitigate core damage.

Industry Operating Experience

The HFE program that supports designing and inspecting the main control room operator interfaces utilizes an extensive operating experience review (Human Factors

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

TR-0915-17772-NP
Rev. 1

Engineering Operating Experience Review Results Summary Report, RP-0316-17614 [Reference 6.6.3]). This is to ensure that best practices and lessons learned are incorporated into the design. Industry experience has also been incorporated into PRA and safety analysis results to ensure the most accurate accident progression modeling is available.

<u>NuScale Operating Experience</u>

NuScale is using the many years of combined experience of previous license holders as input into the design of the controls, procedures, emergency plan, and conduct of operations. This experience is used to influence the operator interface to allow for quick diagnosis and communication of accident conditions.

<u>Simulator and Human-System Interface (HSI)</u>

The simulator runs a high-fidelity model of the thermal-hydraulic characteristics inside the module using RELAP5 and a 3D core model using the reactor core physics modeling software, S3R. This allows NuScale operators to gather the necessary experience and training needed to perform the required duties for the safe operation of the facility. NuScale is developing features of the simulator and the NuScale HSI that aid the operator to perform appropriate actions, including:

- alarm logic that only annunciates when action is required

- procedures integrated into the control interface

- EOPs that are symptom based

- only three safety functions required to monitor for core damage

- all DBAs require no operator action to prevent core damage

E.7 Conclusion

The NuScale design includes multiple barriers to fission product release. The fission product barriers are maintained intact without the need for active cooling, electrical power, external water sources, or operator actions. Multiple passive safety systems must fail, which are events that have a frequency less that 1E-8 per module critical year, in order for damage to occur. Even in the unlikely event of core damage, the time to start of fission product release is longer than the time necessary for the operator to implement mitigating actions. Operators will be trained to recognize the symptoms and take action well before core damage can occur. These attributes of the NuScale design will enhance the ability of the operators to successfully carry out mitigation actions, should such actions be necessary.

Enclosure 3:

Affidavit of Thomas A. Bergman, AF-0218-58900

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman, state as follows:

(1)   I am the Vice President 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 method by which NuScale develops its "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites."

NuScale has performed significant research and evaluation to develop a basis for this method 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 at NuScale Small Modular Reactor Plant Sites".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 March 12, 2018.

_____
Thomas A. Bergman