June 10, 2022

Docket: 99902043

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information (RAI No. 9828) on the NuScale Topical Report, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones," TR-0915-17772, Revision 2

REFERENCES:
1. NRC Letter RAI 9828 - EPZ TR, dated April 22, 2021, RAI# 9828
2. NuScale Topical Report Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones, TR-0915-17772, Revision 2

The purpose of this letter is to provide NuScale's response to NRC Request for Additional Information (RAI), RAI# 9828, noted in the References above. Responses to individual RAI questions are provided in the attached Enclosures.

This letter contains NuScale's response to the following RAI Questions from NRC RAI# 9828:

- 01.05-43
- 01.05-44
- 01.05-45
- 01.05-46
- 01.05-47
- 01.05-48

Enclosures are grouped with all proprietary version responses first, followed by all nonproprietary version responses. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit supports this request. The proprietary enclosures have been deemed to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR § 810.

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

Please contact Liz English at 541-452-7333 or at eenglish@nuscalepower.com if you have any questions.

Sincerely,

Mark W. Shaver

Mark Shaver
Manager, Licensing
NuScale Power, LLC

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

Enclosure 1: NuScale Response to NRC Request for Additional Information RAI# 9828, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information RAI# 9828, nonproprietary

Enclosure 3: Affidavit of Mark Shaver, AF-119894

Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9828, proprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9828, nonproprietary

NuScale Power, LLC
1100 NE Circle Blvd., Suite 200 Corvallis, Oregon 97330, Office: 541.360.0500, Fax: 541.207.3928
www.nuscalepower.com

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Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828
Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-43

The following regulatory basis and discussion applies to all six questions in this request for additional information (RAI).

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

The Commission Policy Statement, "Safety Goals for the Operations of Nuclear Power Plants" (Federal Register Notice 51 FR 28044) established Quantitative Health Objectives (QHOs), which broadly define an acceptable level of radiological risk to the public from nuclear power plant operation. The QHOs have been translated into two numerical objectives:

- the individual risk of prompt fatality from a reactor accident (includes the aggregate of possible reactor accidents) should be less than 5E-7 per reactor-year (ry). The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the plant site boundary;
- the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to 2E-6 per ry. The "area" is understood to be an annulus of 10-mile radius from the plant site boundary.

According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," the COL applicant would perform accident sequence screening based on frequency. First, all external event sequences with an initiating event frequency less than 1E-5 per year would screen out. Second, all internal and remaining external events with a core damage sequence frequency less than 1E-7 per year would screen out before performing consequence analysis.

Request

The staff requests that the applicant justify in the LTR how the methodology confirms that the aggregate of the screened-out sequences do not cause the QHOs to be exceeded or provide revised screening criteria which provides this justification.

NuScale Response:

Emergency planning (EP) is a defense-in-depth feature for mitigating the consequence of radiological releases; a licensee's emergency planning zone (EPZ) does not affect the assessment of risk from an individual nuclear power plant. This is explicitly recognized in the Safety Goal Policy Statement. After proposing the surrogate guideline for the "level of safety ascribed to a plant" as a large release frequency (LRF) of less than 1E-6 per year, the Policy Statement provides that "Furthermore, emergency response capabilities are mandated to provide additional defense-in-depth protection to the surrounding population." As stated in SECY-2013-0029, if the LRF is less than 1E-6 per year, then the quantitative health objectives (QHOs) are met; the LRF guideline is "inherently more conservative than either of the QHOs." Demonstration of LRF necessarily ignores any additional consequence mitigation afforded by EP.

Commission policy is that the Safety Goal objectives are to be applied to all designs, and subsidiary objectives (core damage frequency and containment performance) can be used to implement the LRF objective (SRM-SECY-89-102). Therefore, consistent with the Safety Goal Policy Statement and U.S Nuclear Regulatory Commission (NRC) guidance, NuScale expects that any applicant for a plant license, including one utilizing the EPZ licensing topical report (LTR) methodology, would need to demonstrate that the LRF guideline is satisfied for their design, independent of EP considerations. This expectation is explicit in Section 3.2 of the EPZ LTR, which provides that for a design seeking an NRC approval (e.g., design certification application, standard design approval application, combined license application), an applicant is expected to show that the risk to the public already meets the QHOs in the design application. Therefore, an implicit condition of use for the EPZ methodology is that the plant design is already consistent with the QHOs.

Because any applicant or licensee is expected to satisfy the Safety Goals without consideration of EP, the screened-out sequences under the EPZ methodology cannot cause the QHOs to be exceeded. Any additional effort to show that a plant's EPZ methodology meets the QHOs when the NRC staff have separately concluded that the plant design achieves a total LRF is below

1E-6 per year is unnecessary. Therefore, the EPZ LTR is revised to remove discussion relying on the QHOs to evaluate a subset of plant risk.

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, has been revised as described in the response above and as shown in the markup provided in this response.

Licensing Topical Report

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

~~??? 2021~~

Draft Revision 3

Docket: PROJ99902043

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

Department of Energy Acknowledgement and Disclaimer

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List of Affected Pages

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

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Abstract

The purpose of this licensing topical report (LTR) is to provide the technical basis for NuScale's plume exposure pathway emergency planning zone (EPZ) sizing methodology. The ingestion EPZ is not addressed in this methodology, as the determination of this distance is dependent on land usage that is site-specific. The methodology is informed by the Nuclear Energy Institute (NEI) risk-informed EPZ methodology (Proposed Methodology and Criteria for Establishing the Technical Basis for Small Modular Reactor Emergency Planning Zone, Reference 6.1.5) and extends this risk-informed methodology to address the issue of determining the appropriate accident sequences to be included in the EPZ technical basis, and to consider a consequence orientation in the approach. The screening of accident sequences includes the use of quantitative insights from probabilistic risk assessment (PRA) as RAI 9828 01.05-43, RAI 9828 01.05-44, RAI 9828 01.05-46well as application of engineering insights emphasizing safety margin and layers of defense-in-depth. The screening methodology includes consideration of all hazards and operating modes and also contains integrated assessment of both multi-module effects, when applicable, and uncertainty analysis. Based on the accident sequence screening, the risk results, including source terms and off-site dose versus distance, will serve as the basis for a plume exposure EPZ size. The methodology is intended for use by NuScale~~all advanced nuclear reactor designs, particularly~~ light water small modular reactors (SMRs).~~ such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs.~~

~~The main body of t~~The LTR contains the plume exposure EPZ size methodology for which NRC approval is being sought. In some instances, NuScale design certification application information is used as examples to explain the methodology; however NuScale design information does not form the basis of the methodology.~~ Additionally, 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. Appendices D and E also contain examples of multi-module assessment and operational mitigation features, respectively. NuScale is not seeking NRC approval of the information in the appendices.~~

The topical report requests an NRC review of NuScale's plume exposure EPZ sizing methodology. NuScale also requests, as part of this review and associated comment resolution, that the NRC provide a safety evaluation report ~~(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 applicant, is an acceptable approach for justifying the plume exposure EPZ size.
2. Identification of issues related to the EPZ technical basis that are to be resolved prior to or as part of the application review process.

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

Executive Summary

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

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

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

The main body of the LTR, in Section 3.0, presents the EPZ size methodology for which NRC approval is sought. In some instances, NuScale design certification application information is used as examples to explain the methodology; however NuScale design information does not form the basis of the methodology. The methodology requires compilation of accident sequences from the PRA for all initiators and screens the sequences for inclusion in the EPZ technical basis based upon multiple criteria. ~~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. Appendices D and E also contain examples of multi-module assessment and operational mitigation features, respectively. NuScale is not seeking NRC approval of the information in the appendices.~~

The methodology first determines the ~~appropriate~~spectrum of accident sequences to be evaluated for EPZ. The screening of accident sequences includes the use of quantitative insights from the PRA, including consideration of uncertainty, as well as application of engineering

insights emphasizing safety margin and layers of defense-in-depth. Both "less severe" and "more severe" sequences are evaluated, differentiated by containment status (intact or failed). Based on the accident sequence screening, the risk results, including source terms and off-site dose versus distance, serve as the basis for a plume exposure EPZ size methodology.

The dose criteria for the NuScale EPZ methodology have been defined based on the original EPZ bases, as noted, and the Environmental Protection Agency's (EPA's) protective action guides (PAGs) (PAG Manual, EPA-400/R-17/001, Reference 6.1.4), applied to the sequences as follows: (1) 1 to 5 rem total dose effective equivalent (TEDE) for a design-basis accident (DBA); (2) 1 to 5 rem TEDE for less severe sequences; and (3) 200 rem whole body acute dose for more severe sequences.

Using the risk-informed methodology developed to select appropriate accident sequences, NuScale has also developed a method to evaluate the source term and dose consequence for both less severe and more severe accidents, as presented in Section 4.0. This methodology includes integrated uncertainty analysis. ~~The example results in the appendices, although not subject to approval, indicate that NuScale accident sequences are very infrequent and, even if such accidents occur, would not be expected to produce significant off-site consequences.~~

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

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

1.0 Introduction

1.1 Purpose

The purpose of this licensing topical report (LTR) is to provide a methodology and criteria that can be implemented to establish the design-specific and site-specific plume exposure emergency planning zone (EPZ) size. The methodology is intended for use by NuScale light water~~all advanced nuclear reactor designs, particularly~~ small modular reactors (SMRs). ~~such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs.~~ 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 plume exposure EPZ sizing methodology.

1.2 Scope

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

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

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

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

~~also contain examples of multi-module assessment and operational mitigation features, respectively. The information in the appendices is provided to facilitate: (1) NRC's review of the 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. Rather, the appendices are examples illustrative of applying the methodology.~~ This LTR is not part of the NuScale ~~design certification application (~~DCA~~)~~.

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

1.3 Abbreviations and Definitions

Table 1-1 Abbreviations

Term	Definition
AEF~~AOP~~	annual exceedance frequency~~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~~
DOE	Department of Energy
~~EAL~~	~~emergency action level~~
~~ECCS~~	~~emergency core cooling system~~
EDMG	extensive damage mitigating guideline
~~ELAP~~	~~extended loss of AC power~~
EOP	emergency operating procedure

Table 1-1 Abbreviations (Continued)

Term	Definition
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ESP	early site permit
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
~~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

Table 1-1 Abbreviations (Continued)

Term	Definition
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
TVA	Tennessee Valley Authority
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 result in the loss of a critical safety function, cause any significant damage, nor lead to accident conditions.~~
acute whole body dose	Dose produced by short duration exposure, usually over a matter of minutes, to deeply penetrating radiation (i.e., x-rays, gammas, or neutrons able to reach internal organs) uniformly over a significant portion of the body. Acute whole body dose is represented by dose to red bone marrow and is used to predict the onset of acute radiation syndrome deterministic health effects.
~~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.
emergency planning zone	An area surrounding a plant with a well-defined boundary for which emergency planning is provided, including provisions for protective actions such as evacuation and sheltering.
engineered safety feature	A structure, system, or component that is relied upon during, or following design-basis events to ensure the capability to prevent or mitigate the consequences of those events that could result in potential off-site exposures comparable to the guideline exposures of 10 CFR 50.34(a)(1) (Reference 6.1.9) excluding reactor coolant pressure boundary and reactor protection system items.
exceedance frequency	Annual rate at which a parameter exceeds a given value. As an example, for seismic hazards, the parameter used is the peak ground acceleration.
external hazard	A hazard originating outside a nuclear power plant that directly or indirectly causes an initiating event and may cause safety system failures or operator errors that may lead to core damage or large early release. ~~Events such as earthquakes, tornadoes, and floods from sources outside 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.~~
large release frequency	The frequency of an unmitigated release of airborne fission products from the containment to the environment such that there is a potential for significant radiological doses to the public.

Table 1-2 Definitions (Continued)

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

2.0 Background

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

Protective action zones around commercial nuclear power plants have been an NRC requirement since the early 1960s. Reactor Site Criteria, 10 CFR Part 100 (~~Reference 6.2.1~~Reference 6.2.4) required that every site must have an exclusion area and a low population zone.

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

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

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

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

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

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

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

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

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

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

2.1 Evolution of EPZ-Related Regulatory Requirements and Guidance

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

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

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

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

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

2.2 NEI White Paper

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

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

2.3 NRC EPZ-Related Rulemaking Documents

~~Two recent NRC documents, SECY-15-0077 and SECY-16-0069, address EP-related rulemaking for SMRs.~~In SECY-11-0152 (Reference 6.1.6), the staff informed the Commission of their intent to develop a technology-neutral, dose-based, consequence-oriented emergency planning framework for SMR sites. The staff found that "it may be appropriate for SMRs to develop similarly reduced EPZ sizes, commensurate with their accident source terms, fission product releases, and accident dose characteristics." The projected approach for EPZ sizing was based on accident dose limits at the EPZ boundary, and would include “establishing criteria for determining the point at which the probability of exceeding the PAG is acceptably low.”

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

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

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

In December 2019, the NRC granted an ~~Early Site Permit (~~ESP~~)~~ to the Tennessee Valley Authority (TVA) for the Clinch River Site (CLI-19-10, ~~Reference 6.2.16~~Reference 6.2.17). ~~Part 2,~~ Chapter 13 of the TVA site safety analysis report~~ESP~~ describes a methodology for determining a plume exposure EPZ distance for an SMR installation at the Clinch River Site (Reference 6.2.18). The NRC staff findings summarized in Section C.2 of CLI-19-10 show that the basis for a reduced EPZ size in the TVA ESP is acceptable because it maintains the same level of protection as large ~~light water reactor~~LWR EPZs. Additionally, the Commission "[does] not view TVA's proposal [of site boundary EPZ] as eliminating an element of defense in depth; rather, emergency planning activities would be appropriately scaled to reflect the potential hazards posed by the facility."

In May 2020, the proposed rule for Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities and accompanying draft guidance were released for public comment (Reference 6.2.19 and Reference 6.2.20). The proposed rule would replace the option for case-by-case basis EPZ size determinations for reactors with an authorized power level less than 250MWt with the option to determine the plume exposure pathway EPZ for an SMR or other new technology as the area within which the dose to an individual is projected to exceed 1 rem TEDE over an exposure time of 96 hours from the release of radioactive materials resulting from a spectrum of ~~credible~~ accidents for the facility.

2.4 NuScale Approach

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

Key elements of the NuScale approach include:

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

- A qualitative, plant-level evaluation of defense-in-depth to account for PRA uncertainties.
- A methodology to assess severe accident modeling uncertainties as confirmation of analytical results.

2.5 Conditions of Applicability

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

2.5.1 LWR Condition

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

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

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

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

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

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

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

2.5.2 PRA Conditions

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

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

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

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

3.0 Accident Screening Methodology

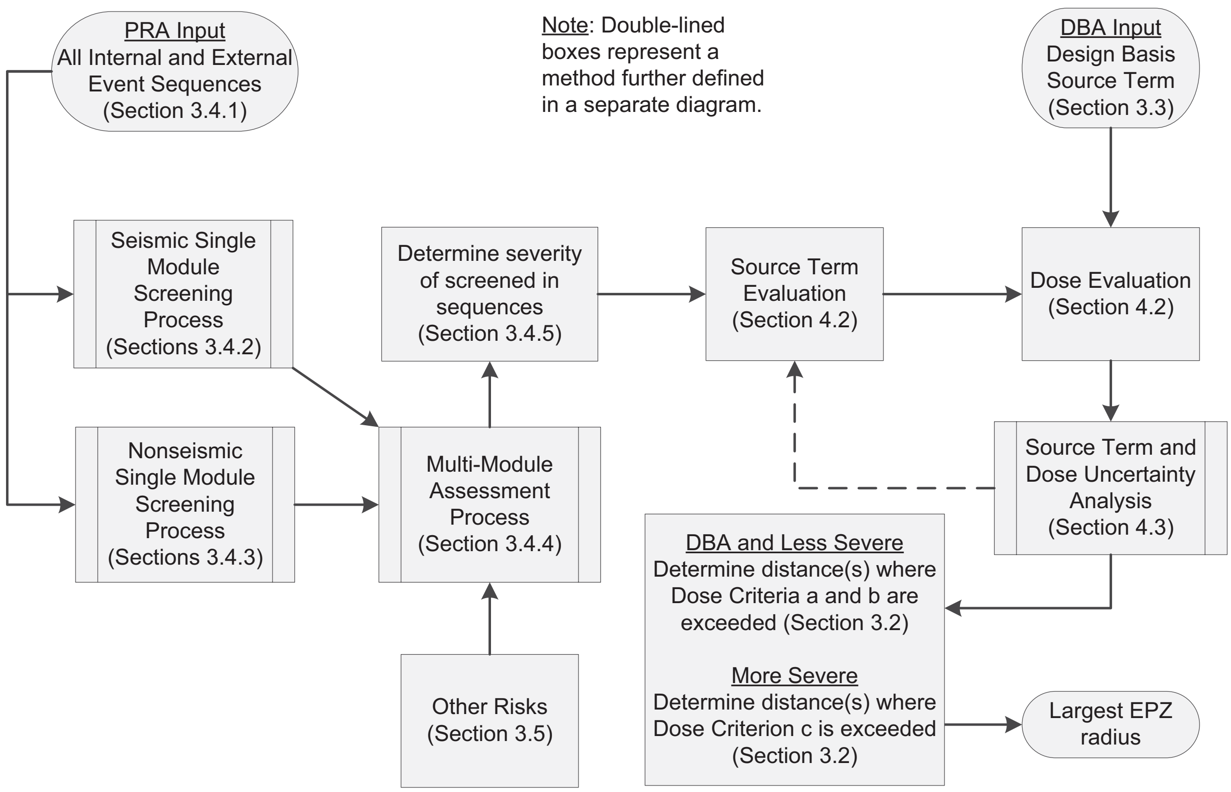
Section 3.0 of the LTR addresses the methodology for determining appropriate accident sequences to be evaluated for the plume exposure EPZ basis. Section 3.1 presents key assumptions and Section 3.2 discusses the dose-based criteria for EPZ sizing. Sections 3.3 and 3.4 discuss the methodology for using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

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

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

    }}2(a),(c) ~~3 for~~ screened-in more severe accident sequences (Section 4.2.3)
  - The site boundary (i.e., the minimum EPZ)

Figure 3-1 Overall methodology to determine EPZ distance



~~An example implementation of portions of the Section 3.4 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.~~

3.1 Assumptions

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

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

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

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

- Evolution of risk-informed applications, including risk-informed changes to the licensing basis for operating plants (i.e., An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant Specific Changes to the Licensing Basis, ~~Regulatory Guide [~~RG~~]~~ 1.174 [Reference 6.3.2]), the reactor oversight process, risk-informed categorization and treatment of structures, systems, and components (i.e., 10 CFR 50.69, ~~Reference 6.3.23~~Reference 6.3.12), and new reactor licensing where all new designs are required to perform a PRA and apply PRA insights to the design process.

- The Risk Management Task Force formed at Commission direction in 2011, culminated with issuance of A Proposed Risk Management Regulatory Framework, NUREG-2150 (Reference 6.3.3), which described a proposed framework for risk management that would provide risk-informed and performance-based defense-in-depth protections.

- The Fukushima Near-Term Task Force issued Recommendations for Enhancing Reactor Safety in the 21st Century (Reference 6.3.4), which recommended establishing a regulatory framework for adequate protection that balances defense-in-depth and risk considerations.

- A risk-informed approach to Emergency Action Levels (i.e., Risk Informing Emergency Preparedness Oversight: Evaluation of Emergency Action Levels-A Pilot Study of Peach Bottom, Surry, and Sequoyah, NUREG/CR-7154, Reference 6.3.5).

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

- SECY-10-0034 (Reference 6.2.10) states that the NRC staff plans to use a risk-informed and performance-based approach that employs deterministic judgment and analysis complemented by PRA information to review design and license applications for SMRs.

- SECY-11-0152 (Reference 6.1.6) states that an appropriate method for addressing EPZ size would involve using a PRA that includes dose assessment, which is based on current insights in severe accident progression.

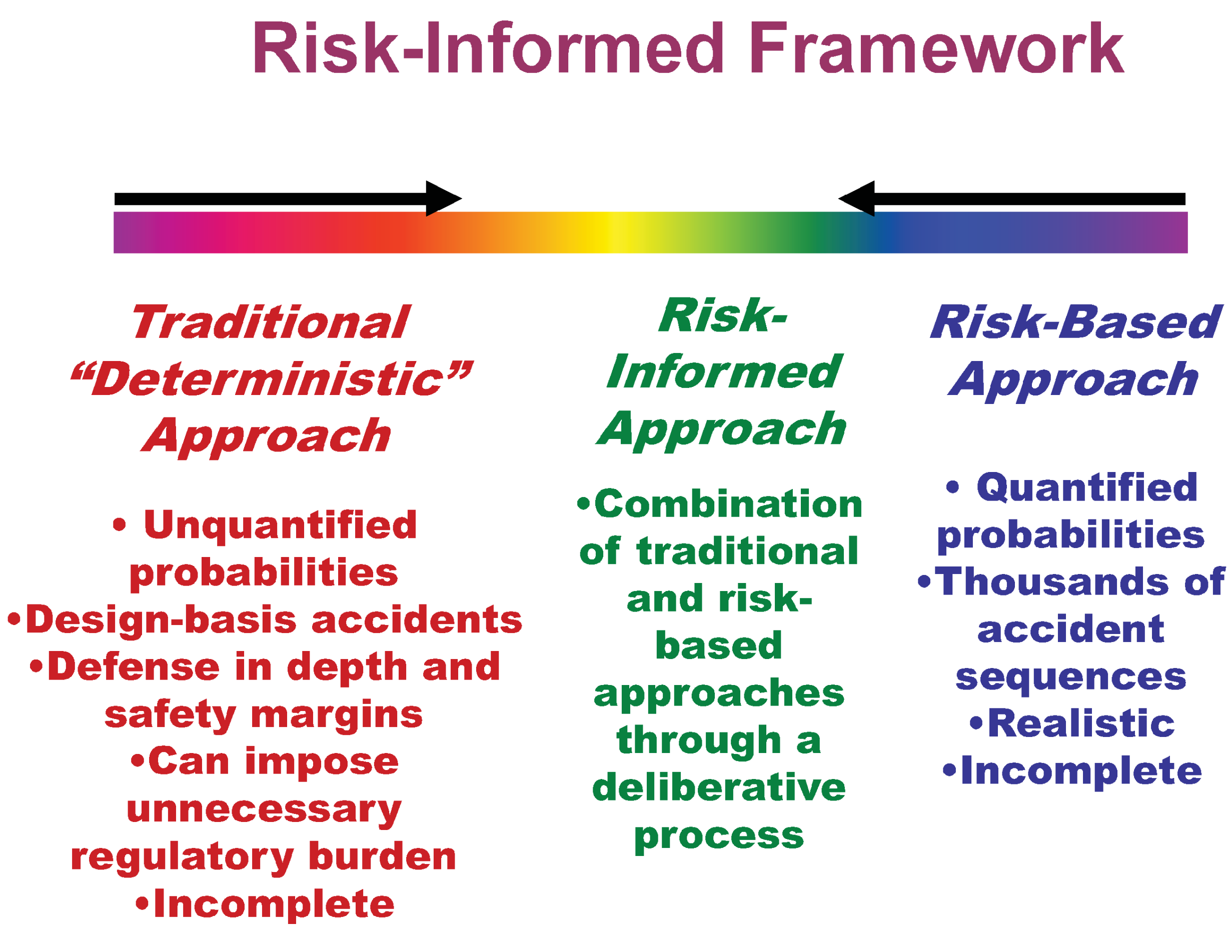
- SECY-15-0077 (~~Reference 6.1.7~~Reference 6.2.3) states that the concept of EPZ size commensurate with the off-site radiological risk is not new to the NRC.

- The recent NRC regulatory basis for SMR emergency preparedness rulemaking (Reference 6.1.10) states that the staff is going to be using a risk-informed approach for selection of licensing basis events and also that the proposed rulemaking will enhance the risk-informed regulatory framework.

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

Justification: Risk-informed processes for any regulatory application, in particular EPZ sizing, should combine and balance insights from deterministic and probabilistic insights. This is illustrated in Figure 3-2, which shows a risk-informed approach as a combination of a traditional, "deterministic" approach, and a risk-based approach, as presented in NRC Risk-Informed and Performance Based Initiatives slides (Reference 6.3.6). A qualitative evaluation of defense-in-depth, consistent with regulatory guidance and practice, is included in the NuScale risk-informed approach to confirm the existence, functionality, and capability of design features and strategies that balance accident prevention and mitigation in order to provide confidence in the acceptably low plant risk and demonstrate protection of the health and safety of the public.

Figure 3-2 Risk-informed framework



Addressing EPZ using a risk-informed approach offers an opportunity to optimize EPZ size and its basis using a more balanced, transparent process. In the generic process used in NUREG-0396 in the 1970s, the margins of safety provided by the EPZ ~~for existing plants were not based on quantification of accidents, but rather~~were based on a combination of risk insights from WASH-1400 and "were qualitatively found adequate as a matter of judgment" (Reference 6.1.8). Given the high level of safety achieved with ~~advanced~~light water SMR designs, as demonstrated by significant margin to the NRC safety goals, this qualitative, generic concept for determining the adequacy of the margins of safety needs to be updated to include a risk-informed, design-specific approach where there is appropriate consideration given to both quantitative and qualitative methods. In the time since NUREG-0396 was published, the severe accident experimental knowledge base and analytical methods have advanced to the point that tools and models are now available to support this methodology for EPZ sizing.

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

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

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

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

- design and operational features that provide multiple, independent layers of defense-in-depth and very low accident sequence frequencies with consideration of uncertainty
- use of state-of-the-art methods to calculate source terms and doses, which greatly reduce uncertainty as compared to previous quantitative methods, that were excessively conservative based on the state of knowledge and methods at that time
  - integrated uncertainty analysis is also used to increase confidence in the best-estimate source term and consequence results as discussed in Section 4.3
- in addition to the state-of-the-art quantitative methods, application of qualitative means to address uncertainties in the context of very low frequency events including:
  - requirement to assess off-site dose consequences from the conservative DBST (Section 3.3)
  - a plant-level, qualitative evaluation of defense-in-depth to demonstrate adequate balance between accident prevention and mitigation of potential consequences as

an extension to emergency planning and NRC's existing defense-in-depth philosophy and guidance

- ~~provision for deterministically-based, operationally-focused mitigation capability as addressed in Appendix E~~
- required site emergency plans will provide a base for expanding response efforts, if necessary, in accordance with regulatory guidance so as to provide an additional layer of defense-in-depth

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

~~Assumption 4: A technically adequate PRA is necessary and acceptable for use in the risk-informed EPZ sizing methodology.~~

~~Justification: Consistent with guidance in the Standard Review Plan, NUREG-0800, Section 19.0 (Reference 6.3.24), the applicant should provide in Chapter 19 of the FSAR an adequate level of documentation to enable the NRC staff to determine the acceptability of the risks to public health and safety associated with operation of a proposed new plant. The staff will confirm the technical adequacy of the PRA is sufficient to support a risk-informed application. As such, applicants who wish to utilize the EPZ sizing methodology should identify use of the PRA to support the risk-informed application in Chapter 19 of the FSAR.~~

~~Applicants need to demonstrate PRA technical adequacy (e.g., RG 1.200 [Reference 6.3.13]). Review of RG 1.200 technical adequacy expectations, however, identifies two obvious areas as not being feasible for a new nuclear power plant design COL application: operational experience and plant walk down. A new design, by definition, will not have operational experience until it has been licensed, constructed, and operated. A plant walk down is not possible if the applicant has not yet constructed the plant. Therefore, for a new design risk-informed application, the applicant will need to demonstrate that the technical adequacy of the PRA is sufficient to support risk-informed decision making.~~

3.2 Dose-Based Criteria

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

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

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

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

{{

}}2(a),(c)

~~Advanced reactor designs are expected to meet with significant margin the NRC core damage frequency and large release frequency safety goals. As such, they would also~~

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

~~meet the NRC quantitative health objectives for protecting the public from undue fatality risks.~~

3.3 Determination of Appropriate Design-Basis Accidents to Be Evaluated

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

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

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

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

~~Figure 3-3 provides a high-level overview of the screening methodology.~~ There are three main elements of the method to identify the spectrum of accidents: (1) initial sequence compilation (described in Section 3.4.1), (2) accident sequence screening ~~based on frequency~~ (described in Sections 3.4.2, 3.4.3, and 3.4.4), and (3) final classification of severity (described in Section 3.4.5).

~~The EPZ methodology requires a site-specific PRA that addresses all internal and external events and hazards and all operating power levels including low power and shutdown.~~ A multi-module process for designs with multiple co-located modules is detailed in Section 3.4.4.

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

3.4.1 Compilation of Probabilistic Risk Assessment Accident Sequences

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

~~The total CDFs from all initiating events and operating modes are summed to calculate the total CDF per module year (used in the probability of dose exceedance calculation, Section 4.2.3).~~ A point estimate sequence-level core damage frequency (CDF), which is calculated to approximate the mean of a distribution, ~~as opposed to a higher percentile value from a PRA uncertainty analysis,~~ is appropriate for a best-estimate EPZ evaluation. The EPZ methodology incorporates a separate uncertainty analysis, which supports the use of a best-estimate evaluation.

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

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

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

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

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

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

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

}}2(a),(c)

{{

}}2(a),(c)

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

}}2(a),(c)

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

{{

}}2(a),(c)

Figure 3-4 {{ }}2(a),(c)

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

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}}2(a),(c)

{{

}}2(a),(c)

- ~~The occurrence frequency of the external event initiator is above 1E-5 per year.~~

~~An initial screening criterion is used for external events: only events with an exceedance frequency at or greater than 1E-5 per year (1 in 100,000 years) are evaluated for potential inclusion in the EPZ size determination based on CDF (Section 3.4.3). Since external hazards are site-specific, the limiting events that correspond to a 1E-5 per year exceedance frequency also vary by site.~~

~~The main justification for selecting a 1E-5 per year external event initiator screening threshold is that it follows the recent precedent granting exemptions for an EPZ of less than 10 miles for sites under decommissioning. In particular, NUREG-2161 (Reference 6.3.19), which is the primary technical basis for a proposed decommissioned plant rulemaking, cited a 1.7E-5 per year occurrence frequency earthquake as "stronger than the maximum earthquake reasonably expected to occur for the reference plant". NRC staff based their decision on the most severe earthquake considered "credible" in drawing its conclusions. While the NUREG basis~~

~~is discussed only with respect to seismic events, it is purely based on frequency and therefore is equally applicable to any external event that can cause fuel damage and off-site consequences. For this methodology, the threshold for what is considered "credible" is conservatively rounded to 1E-5 per year.~~

~~NUREG-2161 further demonstrates that an earthquake of 1.7E-5 per year frequency will produce only moderate leakage in a SFP. A more severe (and lower likelihood) earthquake would drain the SFP more rapidly, but was not included in the consequence analysis as it was not considered "credible". This precedent shows that regardless of the severity of the consequences, very low frequency seismic hazards were not used as the basis for EPZ sizing at decommissioned sites.~~

~~While not the basis for the screening threshold used in the methodology, a relevant factor to consider is the effectiveness of planned actions to respond to an extreme external event. NUREG-1738 (Reference 6.3.20), also in the context of a decommissioned plant, states that "in the large seismic events that dominate SFP risk, pre-planning for radiological accidents would have marginal benefit due to extensive collateral damage offsite. Accordingly, relaxations in EP requirements are not expected to substantially alter the outcome from such a large seismic event." This observation that damage to the surrounding infrastructure would limit emergency planning remains valid irrespective of technical differences between a potential source term from a decommissioned site and from an applicant's design. The underlying regulatory basis of protection of the public health and safety is identical for both potential source terms.~~

~~Interim staff guidance NSIR/DPR-ISG-02 (Reference 6.3.21) provides guidance for the approval of exemption requests for decommissioning plants. One of the criteria stated was to demonstrate a high confidence (95%) of low probability (less than 1E-5 in any given year) of seismic failure. This guidance demonstrates that seismic events with occurrence frequencies below 1E-5 per year were not intended to be used as a basis for approving emergency planning exemptions.~~

~~There is no recorded history of external events below a 1E-5 per year exceedance frequency. While there may be some technical basis for extrapolation, quantification of their likelihood relies on highly uncertain and subjective extrapolation methods (Reference 6.4.20).~~

~~Regardless of the final EPZ distance determined by application of this methodology, a nuclear power plant is required to have an on-site emergency plan to respond to a severe event that can be expanded as necessary. Additionally, while not credited in the methodology or used as the basis for EPZ sizing, local government jurisdictions will develop plans, such as an integrated, all-hazards off-site plan. These plans, although not subject to NRC or FEMA review, would function to mobilize response and prioritize allocation of resources after the occurrence of a very low frequency severe external event. Taken together, in the very unlikely event that a response were necessary, such plans would respond to the infrastructure damage and societal risks that would be much greater than the accident risks from the plant.~~

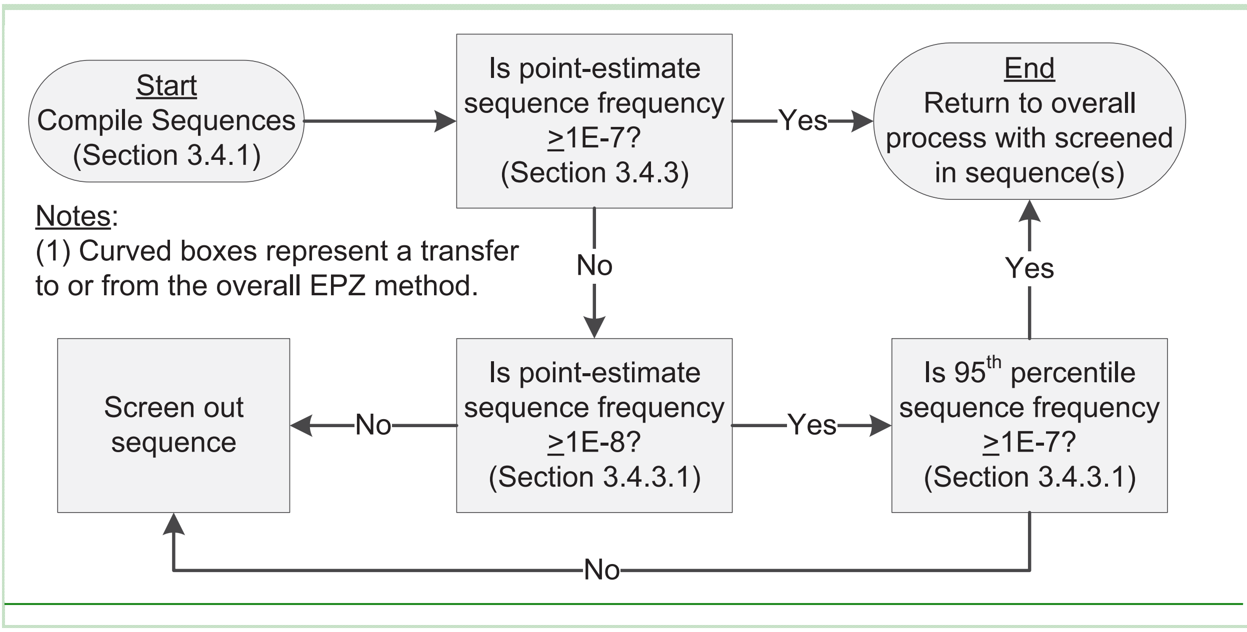
~~The 1E-5 per year initiating event screening frequency applies to all external events for the purposes of EPZ sizing, unless alternate NRC-endorsed criteria exist. It is noted that such a screening criterion would not be used when performing a complete external event PRA for licensing purposes. However, it is an accepted basis for determining emergency planning requirements.~~

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

This section describes the ~~accident sequence~~ screening process for all nonseismic accident sequences. This screening is based on the sequence CDF; all nonseismic core damage sequences with a frequency greater than or equal to~~, including the basis for the~~ 1E-7 per module year are screened into the EPZ sizing analysis.~~screening threshold.~~ Parametric uncertainty in initially screened-out sequences is evaluated to ensure the complete set of source terms is retained in the spectrum of accident sequences. Figure 3-5 provides an overview of the nonseismic sequence screening process. It is noted here that the sequence CDFs that are compared to the screening criteria are to be point estimates that are calculated using mean values for the PRA model basic events. The use of mean values for PRA basic events is typically done in PRA quantification, but the calculated sequence CDF is termed a point estimate since this calculation result will typically differ by a small amount from the mean value CDF that results from a full uncertainty analysis of the PRA. Applicants demonstrate acceptability by:

- screening in all ~~external event severe~~nonseismic core damage accident sequences with~~ an initiating event occurrence frequency ≥ 1E-5 per year (Section 3.4.2) and~~ a point estimate sequence frequency ≥ 1E-7/module year
- evaluating parametric uncertainty in nonseismic core damage accident sequences with a point estimate sequence frequency near the 1E-7/module year threshold~~screening in all internal event severe accident sequences with a frequency ≥ 1E-7/module year~~

Figure 3-5 Nonseismic Sequence Screening Process



All frequencies in this section are per module year to ensure consistency with past precedent of per reactor year in NUREG-0396 (Reference 6.1.3) ~~and SOARCA (Reference 6.2.14)~~. It is also noted that the use of per module year in the screening process does not impact the treatment of multi-module accidents. Multi-module accidents are addressed in Section 3.4.4.

All nonseismic core damage sequences with a frequency greater than or equal to 1E-7 per module year ~~(excluding external events that screen out via the Section 3.4.2 criterion)~~ will be retained for analysis in the EPZ technical basis. This screening criterion is consistent with ~~NRC guidance and previous analyses such as SOARCA (Reference 6.2.14) and Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing, NUREG-1860 (Reference 6.3.8)~~the spectrum of accidents that form the basis for the original EPZ sizing in NUREG-0396, capturing the range of WASH-1400 release category frequencies. Additionally, identifying the spectrum of accident sequences based on CDF conservatively ignores the conditional probability of radionuclide release when compared to the quantification of WASH-1400 release category frequencies. ~~In addition, 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 (ISG) 028 (Reference 6.3.9) lists a set of screening criteria for initiating events ranging from 1E-6 to 1E-8 per reactor year depending on additional system failures.~~

~~The NRC's quantitative health objectives (QHOs), which are described in the Safety Goal Policy Statement (Reference 6.3.10), provide further basis for the 1E-7 per module year sequence frequency screening threshold:~~

- ~~The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.~~
- ~~The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent of the sum of cancer fatality risk resulting from all other causes.~~

~~The prompt fatality rate for the US population, using 2017 data, was 7.59E-4 per year (Reference 6.3.11). The rate for cancer fatalities was 1.53E-4 per year (Reference 6.3.12). Screening at a threshold of 1E-7 per module year is below 0.1 percent of both the prompt and cancer fatality risk from other causes, irrespective of the consequences. Even if the fatality rate to an individual due to an accident occurring at 1E-7 per module year assumed the bounding value of 1.0, it would not exceed one-tenth of one percent of the general prompt or cancer fatality risk to any individual.~~

~~Finally, this threshold is consistent with the lower screening threshold in the TVA ESP plume exposure pathway EPZ methodology, as reviewed by the NRC staff and approved by the Commission in CLI-19-10 (Reference 6.2.18). Section 13.3.3.1 of Part 2 of the recent Early Site Permit granted to the Tennessee Valley Authority (Reference 6.2.18) for a to-be-determined SMR design at the Clinch River Site includes a plume exposure pathway EPZ determination methodology supported by mean accident sequence frequency screening thresholds as low as 1E-7 per year. The EPZ methodology described in this topical report uses a screening threshold consistent with the lower threshold in the approved TVA methodology. Screening thresholds are used to ensure a consequence orientation to the EPZ methodology, consistent with Assumption 3 in Section 3.1.~~

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

~~A screening threshold of 1E-7 per module year represents sequences that occur at approximately one in ten million years. For perspective, per the National Aeronautics and Space Administration, an asteroid impact large enough to degrade the global climate, leading to widespread crop failure and loss of life that would place the entire population of the Earth at risk, is estimated on average to take place several times per million years (Reference 6.3.22). The EPZ methodology considers accidents less likely than this type of asteroid impact.~~

3.4.3.1 Parameter Uncertainty in Nonseismic Sequence Screening

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

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

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

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

1. If the point-estimate value of the sequence is an order of magnitude or more below the screening criteria (i.e., less than or equal to 1E-8 per module year), the sequence is screened-out without further consideration.
2. If the point-estimate value of the sequence is within an order of magnitude of the screening criteria, then the 95th percentile value will be compared to the screening criterion. If the 95th percentile value is also below the screening criterion, then the sequence is screened-out of the process.
3. If the point-estimate value of the sequence is within an order of magnitude of the screening criteria and the 95th percentile value is greater than or equal to the screening criteria, then the sequence is screened-into the process and included in the source term and dose analysis.

3.4.4 Multi-Module Accident Methodology

This section applies to designs that feature multiple co-located reactor modules. Applicants demonstrate acceptability by retaining multi-module accident sequences that meet the following criteria, as shown in ~~Figure 3-6~~Figure 3-6:

- The sequence contains a site-wide or multi-module initiator, or a single-module initiator that physically propagates to other modules.
- Safe shutdown mechanisms are directly compromised in multiple modules, or a coupled safe shutdown function is failed.

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

- The resultant nonseismic multi-module accident sequences have a CDF greater than or equal to 1E-7/module year, including evaluation of parametric uncertainty.

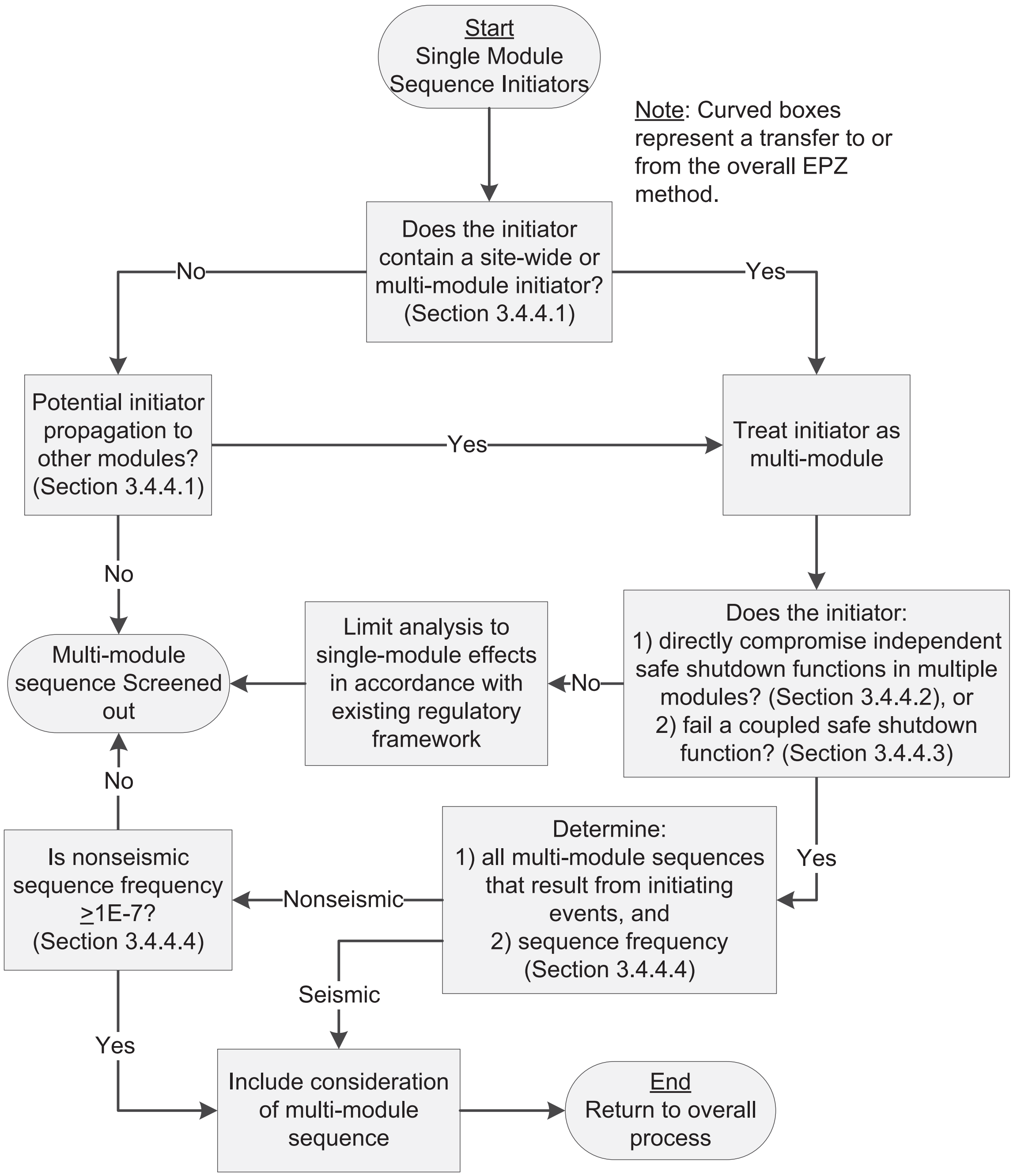
Mechanisms that can cause damage to a single module are analyzed for their potential to propagate to other modules. Only sequences that are screened in by the methodology in Section~~s 3.4.2 and~~ 3.4.3 for a single module are subjected to additional multi-module screening. ~~This is because an accident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

~~Appendix D contains an example assessment, for the NuScale design, 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 3-6:

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

Figure 3-6 Multi-module assessment process



Section 3.4.4.1 describes the multi-module implications associated with initiating events. Section 3.4.4.2 assesses correlated failures between modules.
Section 3.4.4.3 assesses shared system failures. These sections detail the specific considerations that will be evaluated to follow the process shown in ~~Figure 3-5~~Figure 3-6. ~~Additionally, Appendix D contains an example of qualitative assessment following this methodology.~~

3.4.4.1 Hazards and Initiating Events

The evaluation of multi-module accidents requires that all initiators from screened-in accident sequences be assessed against the criteria described in ~~Figure 3-5~~Figure 3-6. These hazards have the potential to include:

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

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

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

3.4.4.2 Correlated Failures between Modules

There is a potential for the initiating event to induce failures that are correlated between modules. For example, there can be a common failure mechanism of identical but independent systems that are demanded for multiple modules or a structural failure caused by the initiating event could impact multiple modules. In accordance with the process described in ~~Figure 3-5~~Figure 3-6, if the initiator can induce any correlated failures of independent systems, any multi-module impacts will be addressed.

3.4.4.3 Shared System Failures

Plant systems may be shared in whole or in part among multiple modules, meaning their failure could have multi-module impacts. In accordance with the process described in ~~Figure 3-5~~Figure 3-6, if the initiator involves a shared system, or impacts a shared mitigating function between modules, any multi-module impacts will be addressed.

3.4.4.4 Multi-Module Conclusions

3.4.4.4.1 Determination of Multi-Module Accidents

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

As a final step, ~~the~~a 1E-7 per module year CDF screening criterion~~criteria from~~ (Section 3.4.3) and consideration of parameter uncertainty (Section 3.4.3.1) is again used to screen nonseismic multi-module sequences ~~out~~ based on their calculated CDFs. This method, as shown in ~~Figure 3-5~~Figure 3-6, is used to consider nonseismic multi-module impacts for EPZ.

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

}}2(a),(c)

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

3.4.4.4.2 Insights

The current regulatory framework for multi-unit and multi-module accidents does not include core damage or large release success criteria categorized by the number of failed units or modules ~~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.

3.4.5 Final Classification of Accidents by Severity

The final step of the methodology is to classify screened-in PRA accident sequences by containment integrity. Appendix I.B.2 of NUREG-0396 separates

~~BDBAs~~beyond-design-basis accidents into two broad consequence categories: less severe accidents and more severe accidents. Applicants demonstrate acceptability by classifying sequences as follows:

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

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

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

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

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

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

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

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

  }}2(a),(c)

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

{{

}}2(a),(c)

3.5 Other Risks

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

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

This report provides acceptance criteria for two such risks, using the second criterion above:~~, that will be common for some advanced reactor designs:~~ SFPs, and severe accident phenomena. Applicants with other risks in addition to the SFPs and severe accident phenomena demonstrated in the following sections shall~~will need to~~ propose acceptance criteria and demonstrate~~demonstration of~~ acceptability for those risks.

3.5.1 Spent Fuel Pool

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

- The time required to boil off SFP inventory is sufficiently long (i.e. >10 hours) that mitigating measures can be implemented to prevent fuel damage.
- Criticality is precluded during refueling operations and storage of spent fuel at maximum capacity, including following design-basis events.
- Leaks from the SFP can be detected and mitigated through monitoring and replenishment systems.

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

3.5.1.1 Spent Fuel Pool Boil-Off

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

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

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

3.5.1.2 Criticality

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

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

3.5.1.3 Leak from Spent Fuel Pool

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

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

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

3.5.2 Severe Accident Phenomena

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

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

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

3.6 Security Events

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

3.6.1 Design-Basis Threats

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

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

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

3.6.2 Beyond-Design-Basis Security Events

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

3.6.2.1 Aircraft Impact

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

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

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

3.6.2.2 Loss of Large Area

Conditions of Licenses, 10 CFR 50.54, Section (hh)(2) (~~Reference 6.3.17~~Reference 6.3.11), 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:

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

Applicants demonstrate compliance with LOLA regulatory requirements by performing an analysis for LOLA using the guidance provided in B.5.b Phase 2 & 3 Submittal Guidance, NEI 06-12 (~~Reference 6.3.18~~Reference 6.3.13).

3.7 Defense-in-Depth Evaluation

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

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

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

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

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

- preserve a reasonable balance among the layers of defense
- preserve adequate capability of design features without an overreliance on programmatic activities as compensatory measures

- preserve system redundancy, independence, and diversity commensurate with the expected frequency and consequences of challenges to the system, including consideration of uncertainty
- preserve adequate defense against potential common cause failures
- maintain multiple fission product barriers
- preserve sufficient defense against human errors
- continue to meet the intent of the plant's design criteria

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

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

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

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

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

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

4.0 Methodology for Source Term and Dose Evaluations

4.1 Application of Software

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

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

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

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

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

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

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

4.1.1 Severe Accident Software and Modeling

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

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

The user of the methodology needs to have reasonable confidence in the severe accident results. Confirmation of results up to core damage with another thermal hydraulic code is one option to provide confidence in the severe accident results. ~~Examples of confirmation calculations are provided in Appendix A and Appendix B for less severe and more severe accident sequences, respectively. These example calculations are not intended for NRC review and approval as part of an SER for this topical report.~~

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

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

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

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

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

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

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

4.1.2 Off-site Consequence Software and Modeling

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

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

Fosmire building wake model in MACCS version 4.1 or later), or ~~C~~confirmation of ATD~~atmospheric transport and dispersion~~ results within 0.31 mi (0.5 km), as described in Section 4.2.4.

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

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

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

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

The *DOSD60.inp* ~~most recent dose conversion factor (~~DCF~~)~~ file included with the ~~latest released~~legacy version 1.13.1 of MACCS should be used for TEDE dose as recommended with caveats by Use of MACCS Dose Coefficient Files to Compute Total Effective Dose Equivalent (Reference 6.4.18). The *DOSD60.inp* files utilize the appropriate tissue weighting factors for International Commission on Radiation Protection publications 26 and 30, as codified in 10 CFR 20.1003. The most recent DCF file included with the latest version of MACCS should be used for acute red marrow dose. These DCF files~~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). Beginning with version 4.1, MACCS implements the Ramsdell and Fosmire building wake and plume meander model (Implementation of Additional Models into the MACCS Code for Nearfield Consequence Analysis, Reference 6.2.19) that has been accepted by the NRC for plume dispersion and transport in the presence of building wakes. No further confirmation of results is necessary if an NRC approved model for plume dispersion in the presence of building wakes (e.g., the Ramsdell and Fosmire model) is used to estimate dose consequences at distances within 0.31 mi (0.5 km) from the source.

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

To provide additional confidence in the MACCS code, a code comparison was performed in Comparison of Average Transport and Dispersion Among a Gaussian, a Two-Dimensional, and a Three-Dimensional Model, NUREG/CR-6853

(Reference 6.4.7) to benchmark MACCS against other ~~atmospheric transport and dispersion~~ATD codes [i.e., NRC's codes RASCAL (Reference 6.4.13) and RATCHET (Reference 6.4.14), and Lawrence Livermore National Laboratory's computer code ADAPT/LODI (Reference 6.4.15)]. RASCAL is the NRC's computer code for rapid emergency response; whereas, RATCHET is a newer related code with upgraded dispersion and deposition modules. ADAPT/LODI is a state-of-the-art, three-dimensional, advection dispersion code. Agreement among the models used in NUREG/CR-6853 was considered acceptable. Although the distances at which the codes were compared in NUREG/CR-6853 extend beyond the region of interest for EPZ sizing (i.e., >10 miles or 16 kilometers), the results provide confidence in the validity of the MACCS code.

4.2 Source Term and Dose Evaluation Methodology

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

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

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

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

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 FSAR Chapter 15 • No credit for additional mitigating features beyond containment	• Apply MACCS dose-in-place evaluation (different from Chapter 15 dose evaluation) • Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used • Dose Criterion a (Section 3.2) with exposure duration of 96 hours

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

Accident Type	Source Term Evaluation	Dose Evaluation
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 additional mitigating design features to calculate fission product holdup and deposition, and fission product release to environment • Consider all operator mitigation actions	• Apply MACCS dose-in-place evaluation • Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used • Dose Criterion b (Section 3.2) with exposure duration of 96 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, credit other mitigation features such as a reactor building, pool, and building sprays, and ~~calculation~~calculate fission product release to the environment • Consider all operator mitigation actions	• Apply MACCS dose-in-place evaluation • Confirm MACCS modeling inside 0.5 km if the legacy area source building wake model is used • Dose Criterion c (Section 3.2) with exposure duration of 24 hours

4.2.1 Design-Basis Accidents

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

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

Source Term Evaluation Methodology

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

Dose Evaluation Methodology

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

- The EPZ boundary dose calculation will apply a methodology similar to that used in the SOARCA study (Reference 6.2.14), which used MACCS state-of-the-art consequence analysis software.
- MACCS input parameters for the applicable site and design-specific source terms will be developed. An example of a site-specific MACCS model can be found in the Surry SOARCA study (Reference 6.4.5).
- Dose Criterion a will be applied. The ~~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 current recommended MACCS output parameter is the EFFECTIVE organ found in the *DOSD60.inp* DCF file (Reference 6.4.18)~~ICRP60ED~~.
- When the wind shift without rotation plume model is used, which is the current best practice, peak dose[5] on the spatial grid is the desired output (Reference 6.4.3, Section 6.19). If a plume model without wind shift is used, peak centerline dose is the desired output.
- Meteorological files with 64 azimuthal sectors should be used. A new meteorological file should be created as discussed in Section 4.1.2.

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

- Stratified random sampling of the meteorological data shall be used to access the meteorological file every hour over the entire year as the starting point for the release.
- An NRC-accepted model for plume dispersion in the presence of building wakes is applied (e.g., the Ramsdell and Fosmire model).
- If the legacy area source building wake model is used, MACCS is applied for distances down to 0.5 km (0.31 miles) from the reactor. For smaller distances, additional steps discussed in Section 4.2.4 are used to address concerns with area source building wake model MACCS results.
- In performing the dose evaluation, the EPZ will encompass those areas in which projected dose from DBAs could exceed the PAGs.
- Site-specific shielding and protection factors can be used with technical justification; otherwise, the conservative values from NUREG-0396 (Reference 6.1.3) (0.7 for groundshine dose, and 1.0 [no shielding] for cloudshine dose and 1.0 [no protection factor] for inhalation dose) should be used.
- Initially, a dose-in-place model will be used with no ad hoc protective actions taken. As a second step, ad hoc, site-specific protective actions may be considered to determine impact on public dose, including relocating people from regions outside the EPZ, after initial calculations without protective actions determine the EPZ distance. When considered, the COL applicant shall apply a site-specific normal and hotspot relocation time, with appropriate justification.

Parameters and the recommended default values to ensure a stationary population and ~~ICRP60ED~~EFFECTIVE dose exposure for the initial 96 hours (345,600 seconds) following plume arrival are shown in Table 4-2. Shielding values (and relocation times for the optional confirmatory calculation) may be updated to reflect site-specific information.

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

MACCS Input	Parameter	Value
Dose Type	~~ICRP60ED~~EFFECTIVE	N/A
Exposure Duration	ENDEMP	345600 s
Cloudshine Shielding	CSFACT	1.0
Groundshine Shielding	GSHFAC	0.7
Inhalation Shielding	PROTIN	1.0
Evacuation	EVATYP	NONE
Normal Relocation Time	TIMNRM	345600 s
Normal Relocation Dose	DOSNRM	1.0E+10 Sv
Hotspot Relocation Time	TIMHOT	345600 s
Hotspot Relocation Dose	DOSNRM	1.0E+10 Sv

4.2.2 Less Severe Accidents

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

- All accident sequences determined to be less severe accidents in Section 3.4.5 are compared against Dose Criterion b, considering:
  - site-specific meteorology and shielding factors
  - cloudshine, groundshine, inhalation, and resuspension dose pathways
  - no protective actions
  - 96 hour exposure duration following plume arrival
  - ~~ICRP60ED~~EFFECTIVE DCFs~~ dose conversion factors~~

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

The methodology for performing the source term and dose evaluations for less severe accidents is contained within this section.~~ 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 demonstrate application of the methodology, but is not intended to be the basis for a site-specific plume exposure EPZ size.~~

Source Term Evaluation Methodology

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

- Development and benchmarking of the primary severe accident (e.g., MELCOR) model as discussed in Section 4.1. This MELCOR module model is used to calculate radionuclide release fractions from containment, as a function of time, for the less severe accident sequences that were determined as appropriate for evaluation to support EPZ.
- The impact on source term of operationally-focused mitigation (i.e., SAMGs, EDMGs, and other EPZ-oriented operator mitigation actions in addition to EOPs) may be considered.~~ 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 conservative assumption of a small opening sized to result in design-basis leakage at containment design pressure (i.e., the technical specification limit). Alternatively, the fission product release to the environment from a containment modeled as perfectly sealed may

be calculated by multiplying the airborne radionuclide concentration in containment as a function of time by the design-basis leakage rate at containment design pressure.~~ (as performed in Appendix A).~~

- A model of additional fission product barriers external to the containment may optionally be used to calculate fission product deposition and holdup following leakage from the containment. The source term to the environment is then used as input to the dose evaluation.

Dose Evaluation Methodology

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

4.2.3 More Severe Accidents

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

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

Section 3.4 also addresses the methodology for determination of more severe appropriate accident sequences to be evaluated as part of the basis for EPZ size. For any more severe accident sequences that are screened in to the EPZ technical basis source term and dose evaluations will be performed in accordance with the methodology. Dose Criterion c {{ }}[2(a),(c)] is applied.

The methodology for performing the source term and dose evaluations for more severe accidents is contained within this section.~~ Example source term and dose calculations to illustrate the evaluation process for more severe accident sequences are provided in Appendix B. This example is intended to demonstrate application of the methodology, but is not intended to be the basis for a site-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 models of additional external barriers, a separate effects model should be used to credit fission product deposition in the release pathway, such as piping, between the containment and the additional external barriers.

Dose Evaluation Methodology

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

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

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

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

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

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

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

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

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

}}2(a),(c) ~~An example of this secondary presentation of results is found in Appendix B.~~

In equation form {{

}}2(a),(c)

{{

}}2(a),(c)

A simple numerical example illustrating the steps {{

}}2(a),(c) In this example, the EPZ distance would be approximately 0.03~~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~~	~~1.00E-07~~	~~8.60E-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~~	~~9.88E-02~~
~~2~~	~~0.03~~	~~4.00E-02~~	~~7.50E-01~~	~~1.00E+00~~	~~9.24E-02~~
~~3~~	~~0.06~~	~~2.00E-02~~	~~6.00E-01~~	~~9.00E-01~~	~~6.40E-02~~
~~4~~	~~0.13~~	~~0.00E+00~~	~~8.00E-02~~	~~6.00E-02~~	~~5.35E-03~~
~~5~~	~~0.2~~	~~0.00E+00~~	~~0.00E+00~~	~~4.00E-04~~	~~4.65E-06~~

Table 4-5 {{ }}2(a),(c)

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

Figure 4-1 {{

}}2(a),(c)

{{

}}2(a),(c)

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

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

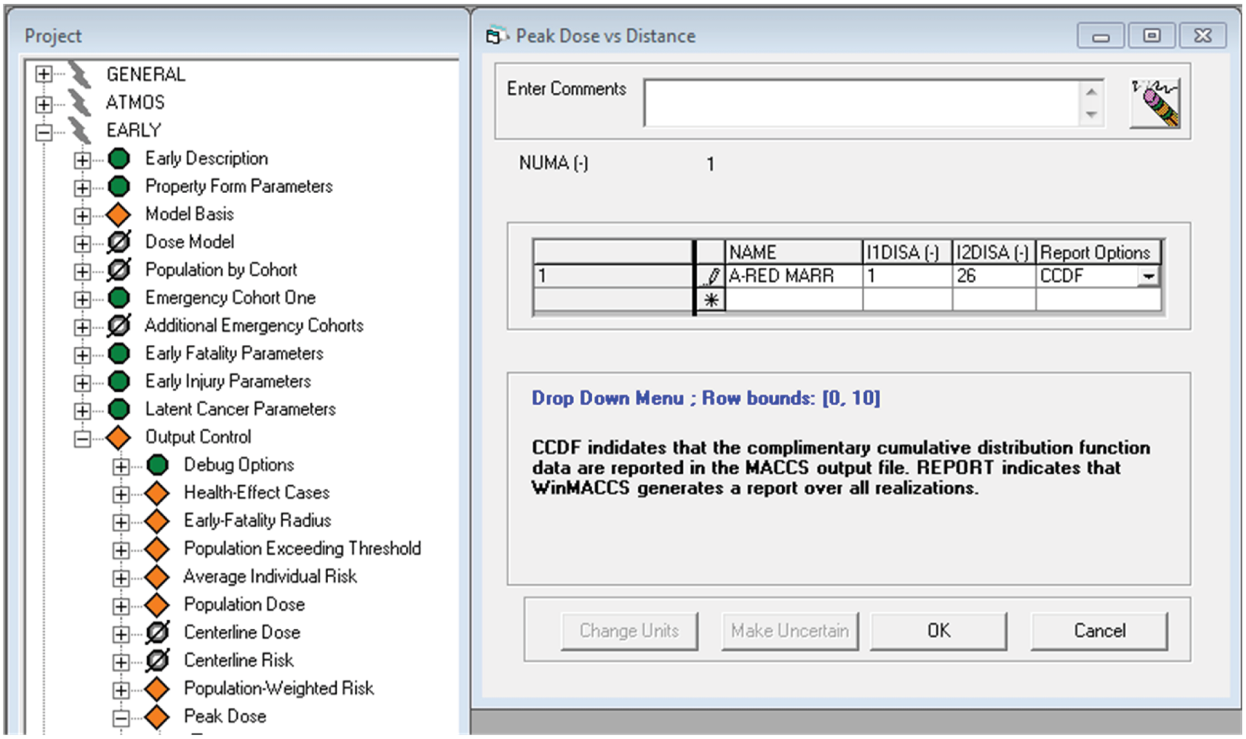


Figure 4-3 Example ~~CCDF~~complementary cumulative distribution function output for a single distance

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

   PEOPLE FRACTION =      1.0000

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

4.2.4 Additional Steps for Dose Evaluation Inside 0.5 Kilometers

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

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

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

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

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

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

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

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

4.3 Uncertainty Analysis Methodology

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

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

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

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

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

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

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

The sequence selection is then followed by the identification of uncertainty parameters for MELCOR and MACCS models that have a significant impact on the off-site radiological releases and the dose consequences. Both epistemic and aleatory parameters should be considered. The parameters from the SOARCA uncertainty analyses should be used as a starting point.~~ Section C.2 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 analyzed design, or for any new parameters, distributions will be technically justified and, when possible, based on physical bounds. Along with the parameters varied, it is also recommended that a table be included with an application that details all parameters considered but ultimately rejected, along with the technical justification for non-inclusion.

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

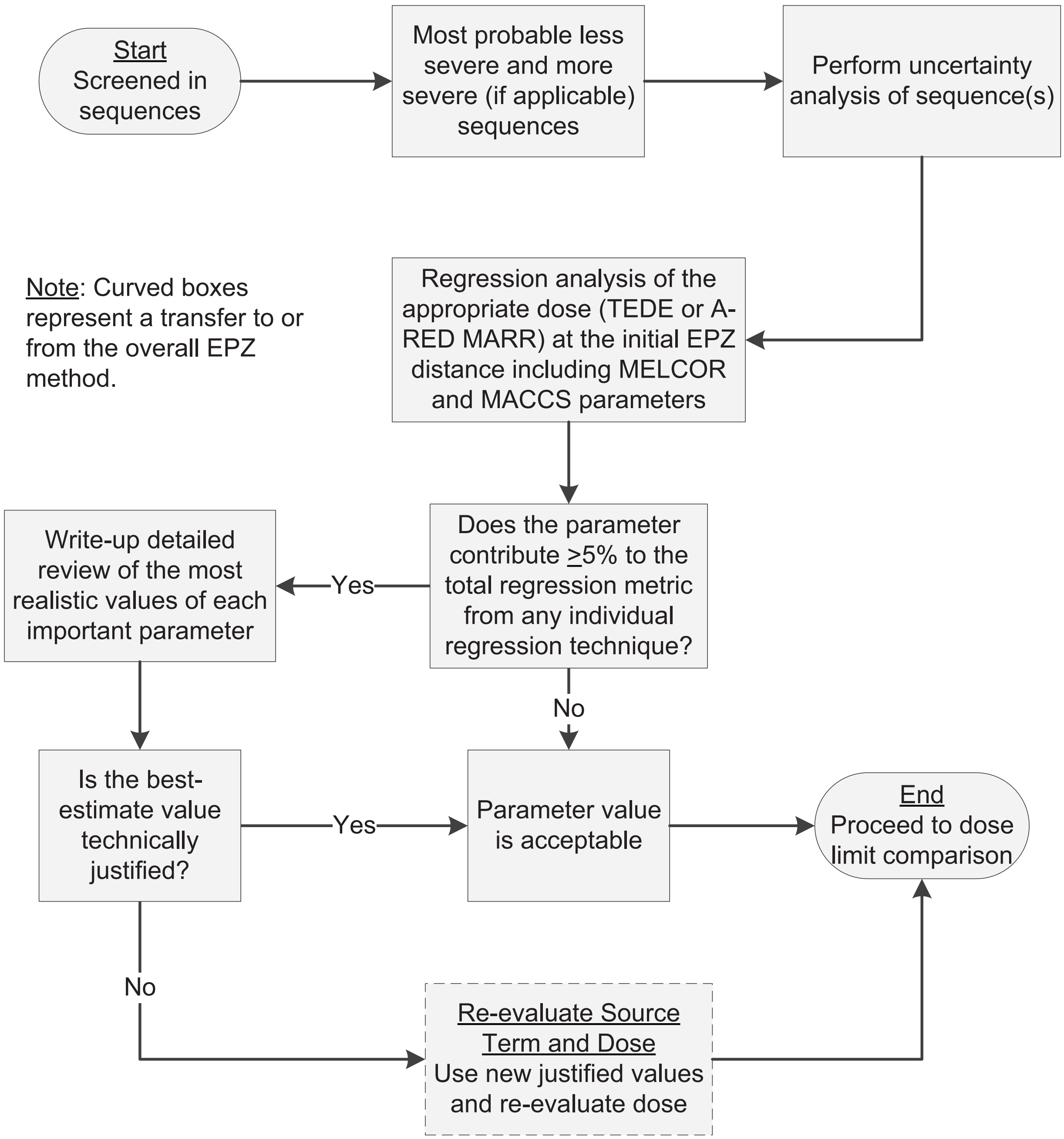
In Step 5, uncertainty analyses are performed on the results after all MELCOR and MACCS runs are completed. The uncertainty analysis is performed with one or more regression techniques (i.e., rank regression, quadratic regression, recursive partitioning, and multivariate adaptive regression splines~~ [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.

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

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

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

Figure 4-4 Application method of uncertainty results



5.0 Summary and Conclusions on Methodology

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

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

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

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

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

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

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

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

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

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

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

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

~~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 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.~~

6.0 References

6.1 Abstract and Section 1.0 Document References

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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.

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6.2 Section 2.0 Document References

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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).

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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. However, these selected accidents are not necessarily the same as would be selected using the Section 3.4 methodology. The analyses of these accidents are solely meant to be an example of application of NuScale's EPZ methodology. NuScale is not requesting NRC approval of these examples. 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.~~

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~~

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

~~{{~~

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

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

~~{{~~

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

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

~~{{~~

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

~~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:~~

- ~~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 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 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).~~

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

{{

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

{{

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

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

~~{{~~

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

~~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.~~

Figure A-6 ~~Containment pressure versus time, MELCOR~~

~~{{~~

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

~~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.~~

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

~~{{~~

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

~~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.~~

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~~

~~{{~~

~~}}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~~

~~{{~~

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

~~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.~~

~~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.~~

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)~~

~~{{~~

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

~~* 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.~~

~~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.~~

~~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.~~

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

~~{{~~

~~}}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)~~

~~{{~~

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

~~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.~~

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.~~

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. NuScale is not requesting NRC approval of these examples.~~

~~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).~~

~~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 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.~~

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)[a]~~

~~{{~~

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

a. ~~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.~~

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

{{

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

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

{{

}}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**~~

~~{{~~

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

~~Pressure-Induced Steam Generator Tube Failure (Unisolated Secondary Line Break) - Base Case~~

~~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.~~

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)[a].~~

~~{{~~

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

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

~~{{~~

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

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

~~{{~~

~~}}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**~~

~~{{~~

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

~~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.~~

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**~~

~~{{~~

~~}}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**~~

~~{{~~

~~}}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~~

~~{{~~

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

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.~~

~~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.~~

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

~~{{~~

~~}}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~~

~~{{~~

~~}}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**~~

~~{{~~

~~}}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**~~

~~{{~~

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

~~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.~~

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~~

~~{{~~

~~}}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~~

~~{{~~

~~}}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**~~

~~{{~~

~~}}~~ ~~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~~

~~{{~~

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

~~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.~~

~~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).~~

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~~

~~{{~~

~~}}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~~

~~{{~~

~~}}~~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**~~

~~{{~~

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

~~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.~~

- ~~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.~~

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)~~

~~{{~~

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

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.~~

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 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. It is important to note that the NuScale DCA PRA contained a seismic margins assessment, which does not produce a CDF result; therefore, seismic sequences are not included in this appendix. Additionally, although some external events are included in this example, the initiating event screening from Section 3.4.2 is not demonstrated. As external events are site specific, there would be limited value in applying this part of the method on generic results. 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 (or roughly 1 in 11 million years).~~

~~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)~~

Table C-1 ~~PRA sequences with CDF contributions~~

~~{{~~

~~}}2(a),(c)~~

Table C-1 ~~PRA sequences with CDF contributions (Continued)~~

~~{{~~

~~}}2(a),(c)~~

C.2 ~~Example Application of Uncertainty Analysis Method~~

C.2.1 ~~Selection of Accident Sequence~~

~~{{~~

~~}}2(a),(c)~~

~~{{~~

~~}}2(a),(c)~~

C.2.2 ~~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)~~

~~{{ }}2(a),(c) 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-2. 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.~~

Table C-2 ~~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~~

Table C-2 ~~Examples of parameters considered for EPZ sizing uncertainty analysis (Continued)~~

~~**MELCOR**~~	~~**MACCS**~~
~~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~~	

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 a NuScale design-specific example of the information that would support multi-module screening as presented in Section 3.4.4. NuScale is not requesting NRC approval of these examples.~~

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 3.4.4.1. 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 is module specific and there is no coupling between safe shutdown functions in multiple modules.~~
~~CVCS LOCA Outside Containment Charging Line~~	
~~CVCS LOCA Outside Containment Letdown Line~~	
~~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 3-6).~~
~~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~~	

Table D-1 ~~Multi-module implications of internal initiators (Continued)~~

~~Initiator~~	~~Multi-Module Implications~~
~~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 and RPV 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.~~

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.~~

Table D-2 ~~Multi-module implications of induced fire events (Continued)~~

~~Category~~	~~Comments~~	~~Multi-Module Implications~~
~~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.~~

~~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 3.4.4.1, 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:~~

- ~~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 3.4.4.1 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 screened based on the steps described in Section 3.4.2.~~

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~~	~~Module specific~~
~~Bioshield - pool wall bolt failure - normal operation~~	~~Shear failure of pool wall anchor bolts~~	~~Core damage/large release~~	
~~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.~~

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.~~

Figure D-1 ~~Inter-module dimensions in reactor pool~~

~~{{~~

~~}}2(a),(c)~~

~~The rows of opposing modules as well as the module transit path are a design-specific NuScale feature.~~

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~~

~~{{~~

~~}}2(a),(c)~~

~~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.~~

~~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.~~

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. NuScale is not requesting NRC approval of this example.~~

~~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, and~~

- ~~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 thermally 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~~

~~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 responding to 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.~~

~~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,~~
- ~~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, and~~
- ~~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.~~

~~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~~

Table E-1 ~~Post-accident monitoring variables (Continued)~~

~~**Safety Function**~~	~~**Major Post-Accident Monitoring Parameters**~~
~~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."~~

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. An example of a FLEX capability is the use of a portable pump to makeup to the SFP.~~

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 un-isolate and initiate CVCS injection~~	~~Local un-isolation 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:~~

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 (Reference 6.2.14).~~

~~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 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.~~

~~NuScale Operating Experience~~

~~NuScale is using the many years of combined experience of previous nuclear power plant 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.~~

~~Simulator and Human-System Interface (HSI)~~

~~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-7 per module 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.~~

Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828
Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-44

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening

Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," (ADAMS Accession No. ML17317A256) states that "all plant operating modes and hazard groups be addressed when those risk contributions affect the decision." The ASME/ANS PRA standard addresses internal flood, internal fire, seismic, wind, external flood and other external hazards. The 1995 PRA Policy Statement states: "PRA evaluations in support of regulatory decisions should be as realistic as practicable and appropriate supporting data should be publicly available for review." In the Staff Requirements Memorandum (SRM) to SECY-04-0118, "Plan for the Implementation of the Commission's Phased Approach to Probabilistic Risk Assessment Quality," states that "the licensee's submittal is expected to be in conformance with the published standards."

According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," the COL applicant would first screen external event sequences with an initiating event frequency less than 1E-5 per year. In LTR Table 1-2, "Definitions," an external event is defined as 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. Internal fires from sources inside and outside the plant are also considered to be external events.

Based on the guidance and expectations outlined above, the staff would expect external and internal events to be screened based on equivalent screening criterion consistent with the 1995 PRA policy statement, SECY 04-0118, and RG 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," Revision 3 (ADAMS Accession No. ML17317A256).

Request

Therefore, the staff requests that the applicant justify in the LTR:

(a) the technical basis for screening external events using a different quantitative screening criterion from internal events given that the ASME/ANS PRA standard addresses internal flood, internal fire, seismic, wind, external flood and other external hazards.

(b) why the methodology is not screening potentially risk significant external events given that external event initiators with a frequency less than 1E-5 per year could have core damage sequence frequencies that result in the QHO's being exceeded.

For example, an external event with an initiating event frequency of 5E-6 per year that causes core damage 50 percent of the time, has a resulting core damage sequence frequency of 2.5E-6 per year. The staff understands that the methodology in the LTR would allow this sequence to be screened out using the screening for external event initiators although it may challenge the large release frequency (LRF) Commission goal for new reactors and, consequently, the QHOs. As another example, regarding the external event screening initiator frequency of less than 1E-5 per year, in RG 1.76, "Design-Basis Tornado and Tornado Missiles for Nuclear Power Plants," Revision 1 (ADAMS Accession No. ML070360253), the design-basis tornado is based on an annual exceedance frequency of 1E-7 per year, and, therefore, is not screened from deterministic analysis.

(c) As an alternative to (a) and (b) above, the applicant may provide revised external event screening criteria with technical justification.

NuScale Response:

(a) Alternative screening criteria are provided. Refer to response (c).

(b) Alternative screening criteria are provided. Refer to response (c).

(c) NuScale is implementing the following changes to the emergency planning zone (EPZ) licensing topical report (LTR) to address external events screening:

- Limit the scope of the LTR to the NuScale light water small modular reactor (SMR) family of designs through a condition of applicability added to Section 2.5.

- Remove the 1E-5 per year annual exceedance frequency (AEF) screening on external event initiating events in Sections 3.4.2 and 3.4.3.
- Internal events and nonseismic external events are screened once at 1E-7 per module year sequence frequency per Section 3.4.3.
- Uncertainty in nonseismic sequences is evaluated consistent with NUREG-1855 by retaining for source term and dose evaluation screened out nonseismic sequences with a 95th percentile frequency greater than or equal to 1E-7 per module year.
- {{

  }}2(a),(c)
- Seismic multi-module sequences are not screened following identification using the process documented in Section 3.4.4 of the LTR.
- {{

  }}2(a),(c)

These methodology modifications support the identification of a spectrum of accidents for consideration in the EPZ sizing basis while maintaining the risk to the public outside the EPZ boundary sufficiently low and consistent with the basis for the original 10 mile EPZ in NUREG-0396.

Limitation of Applicability to the NuScale Design

The revised screening criteria for external events reflect the understanding of accident progression and very low external event risk in the robust NuScale family of designs. Therefore, the EPZ LTR methodology is limited to designs with the following characteristics:

- The light water reactor (LWR) is a NuScale SMR design, including the standard plant design (Docket 52-048) and variations and derivatives thereof comprising all the following characteristics:
  - small modular integral pressurized light water reactors, meaning reactor modules composed of a reactor core, primary cooling loop, pressurizer, and steam generator(s) within a reactor vessel, housed within a containment vessel normally operated at subatmospheric pressure conditions,

- operating modules partially immersed in water that serves as the ultimate heat sink (UHS),
- the UHS is retained below grade in a structure with up to 12 reactor modules per UHS,
- a safe shutdown earthquake with a peak ground acceleration of 0.5g, and
- structures, systems, and components (SSCs) capable of performing their safety functions without AC electric power, DC electric power, or operator actions for at least 72 hours following a design basis event.

Consideration of Nonseismic Risk

Nonseismic probabilistic risk assessment (PRA) sequences are screened at a sequence level frequency of 1E-7 per module year. The 1E-7 per module year threshold is consistent with the spectrum of accidents evaluated in NUREG-0396, capturing the range of WASH-1400 release category frequencies as discussed in Section 3.4.3 of the LTR. Seismic sequences are screened using a separate method as discussed later in this response. Nonseismic external events PRA sequences and internal events PRA sequences have similar characteristics in the NuScale design. Because the modules themselves and SSCs relied upon for safe shutdown are housed in the robust seismic category I reactor building, these SSCs are protected from direct failure due to nonseismic hazards. Nonseismic external events sequences contain random failures of equipment following the external initiating event similar to internal events sequences. Therefore, all nonseismic sequences are evaluated equivalently in the EPZ methodology.

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

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

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

- If the point-estimate value of the sequence is an order of magnitude or more below the screening criteria (i.e., less than or equal to 1E-8 per module year), the sequence is screened-out without further consideration.
- If the point-estimate value of the sequence is within an order of magnitude of the screening criteria, then the 95th percentile value will be compared to the screening criterion. If the 95th percentile value is also below the screening criterion, then the sequence is screened-out of the process.
- If the point-estimate value of the sequence is within an order of magnitude of the screening criteria and the 95th percentile value is equal to or greater than the screening criteria, then the sequence is screened-into the process and included in the source term and dose analysis.

Consideration of Seismic Risk

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

{{

}}2(a),(c)

{{

}}2(a),(c)

Multi-module seismic sequences are identified following the current process in Section 3.4.4 of the LTR. {{

}}2(a),(c)

Dose Criterion c

{{

}}2(a),(c)

{{

}}2(a),(c)

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, has been revised as described in the response above and as shown in the response to RAI Question 01.05-43.

Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828
Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-45

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening

Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

The 1995 PRA policy statement, Draft Guide 1350, "Emergency Planning for Small Modular Reactors (SMRs) and Non Light Water Reactors (ANLWRs)," (ADAMS Accession No.ML18082A044) and NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decisionmaking," Revision 1, Final Report (ADAMS Accession No. ML17062A466), provide the expectation and guidance for the treatment of uncertainties in a risk-informed application.

Request

The staff requests that the applicant address in the LTR how numerical uncertainties associated with each screened core damage sequence are to be considered when comparing against the numerical screening thresholds. Consistent with NUREG-1855, "Guidance on the Treatment of Uncertainties Associated with PRAs in Risk-Informed Decision Making," Revision 1 (ADAMS Accession No. ML17062A466), the applicant should address the following:

(a) The impact of parameter uncertainty on the screening results.

(b) A description of the relevant sources of model uncertainty and their impact on the screening results.

(c) A description of any significant modeling assumptions and their impact on the screening results.

The treatment of uncertainty in the LTR should consider that a COL applicant will not have operating procedures, operating experience (especially for new design features), or the ability to perform walkdowns.

NuScale Response:

Addressing uncertainty is a key element of a technically adequate probabilistic risk assessment (PRA). Uncertainty requirements are included in each part of the ASME/ANS PRA standard and further clarified in DC/COL ISG-028 for design stage considerations, and NUREG-1855 provides guidance on how to treat uncertainties associated with PRA in risk-informed applications. PRA technical adequacy is discussed in the response to question 01.05-46.

As stated in DC/COL ISG-028, when general design and guidance information and general industry practice is used, the PRA will contain more inherent assumptions and increased uncertainty. As a result, the applicant should document the limitations and impacts on the use of the risk results and insights and should document the sources of uncertainty and assumptions resulting from the use of general operational information. In the individual detailed tables, Tables 2 through 9 of DC/COL ISG-028, that address the supporting requirements for each part of the PRA Standard, new supporting requirements are included for each of the technical elements to capture the documentation of the assumptions, uncertainties, and their impacts on the use of the risk results and insights due to the status of the design, site, operational, and maintenance information or data. The PRA used to support the emergency planning zone (EPZ) methodology would include documentation of the additional sources of uncertainty and related assumptions resulting from the status of the design, site, operational, and maintenance information or data. As described in the response to question 01.05-44, Section 3.8 has been added to the LTR to ensure that the key assumptions, modeling uncertainties, and completeness uncertainties in the underlying PRA are reviewed and addressed in the context of EPZ sizing.

Additionally, the impact of uncertainties in the application of the PRA on the sequence screening are addressed through quantitatively evaluating uncertainty against the screening thresholds and complementary deterministic aspects of the EPZ method. Regardless of the results of the sequence screening, Section 3.3 of the licensing topical report (LTR) requires that the offsite design basis source term (a surrogate less severe accident) is evaluated for offsite dose, and Section 3.7 of the LTR requires that a plant level defense-in-depth analysis is performed. Thus, the final EPZ distance determined by the methodology is always based on some dose consequence representative of the design and the demonstration of maintained defense-in-depth, rather than the screening out of all risk. As described in the response to question 01.05-44, {{

}}2(a),(c)

Section 3.4.3.1 is added to quantitatively address parametric uncertainty in the comparison of nonseismic sequence frequencies against the 1E-7 per module year sequence screening threshold, consistent with NUREG-1855.

The treatment of lack of available information at the design stage, such as plant walkdowns or operating experience, is discussed in response to question 01.05-46.

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, has been revised as described in the response above and as shown in the response to RAI Question 01.05-43.

Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828

Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-46

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening

Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

DC/COL ISG-028, "Assessing the Technical Adequacy of the Advanced Light-Water Reactor Probabilistic Risk Assessment for the Design Certification Application and Combined License Application," (ADAMS Accession No. ML16130A468), along with RG 1.200, " An Approach for Determining the Technical Adequacy of Probabilistic Risk Assessment Results for Risk-Informed Activities," Revision 3 (ADAMS Accession No. ML20238B871), provides an acceptable approach to support certification and licensing of reactors under 10 CFR Part 52. It further states that other applications, including risk-informed applications, "need to directly address the application-specific regulations and guidance, including the evaluation of the technical adequacy of the PRA needed for the specific application using the PRA Standard, as endorsed by RG 1.200."

Request

Therefore, the staff is requesting that the applicant address in the LTR:

(a) The need for PRA to be peer reviewed in accordance with NEI 17-07, Revision 2 (for Advanced Light Water Reactors (ALWRs)) (ADAMS Accession No. ML19231A182).

(b) The need for the COL applicant to evaluate hazards/modes where NRC-endorsed Standards do not exist to justify the technical adequacy of the PRA to support the PRA sequence screening.

(c) The need for the PRA to be developed using RG 1.200 for Capability Category II.

(d) The need for the COL applicant to identify and justify any exceptions (e.g., inability to perform walkdowns).

NuScale Response:

(a) Section 2.5, Conditions of Applicability, of the emergency planning zone (EPZ) licensing topical report (LTR) requires probabilistic risk assessment (PRA) acceptability for use of the PRA in the risk-informed EPZ methodology.

PRA acceptability is an important part of using PRA in a risk-informed activity. As such, PRA acceptability was identified as a requirement in the EPZ LTR. The applicant is required to develop documentation of the PRA model and analysis performed to support the EPZ methodology; one approach[[1]] for determining whether a PRA is sufficient to be used in risk-informed regulatory decisionmaking is described in Regulatory Guide (RG) 1.200.

RG 1.200 defines PRA acceptability in terms of an acceptable base PRA, conformance with national consensus standards, and the peer review process. In addition, NEI 17-07 provides guidance for conducting and documenting a PRA peer review using the ASME/ANS PRA standard. As such, an applicant can demonstrate PRA acceptability when implementing the risk-informed EPZ methodology via RG 1.200 and the peer review process described in NEI 17-07. As stated in Section 1-6.1 of the ASME/ANS PRA standard, “The peer review need not assess all aspects of the PRA against all requirements in the Technical Requirements Section …; however, enough aspects of the PRA shall be reviewed for the reviewers to achieve consensus on the adequacy of methodologies and their implementation for each PRA element.” The set of key review areas identified in Sections 1-6.3 and 1-6.6 of the ASME/ANS PRA standard for the technical element(s) being peer reviewed must be addressed.

(b) Section 2.5 of the LTR specifies that the EPZ methodology requires a PRA that addresses internal and external events, and all operating modes. Therefore, an applicant will compile accident sequences for internal events and external events, and all operating modes to support PRA sequence screening in the EPZ methodology.

As described in paragraph (a), PRA acceptability is required for use of the PRA in the risk-informed EPZ methodology. As such, applicants are required to document the bases for determining PRA acceptability for use in the EPZ methodology.

In cases where an applicant uses national consensus standards, but there is not a current U.S. Nuclear Regulatory Commission (NRC) endorsed PRA standard for a specific hazard or mode, applicants need to document the bases for why the method employed is acceptable to support the risk-informing decision making in the EPZ application. Similar justification is needed when an applicant uses a non-PRA-type evaluation. This guidance is consistent with other PRA applications; for example, RG 1.201 states that submittals that are received before the NRC endorses all applicable PRA standards (e.g., shutdown PRAs) are expected to document the bases of why the method employed is acceptable to support the application.

Paragraphs a and b are also consistent with Draft Guide (DG)-1350, Performance-Based Emergency Preparedness for Small Modular Reactors, Non-Light-Water Reactors, and Non-Power Production or Utilization Facilities; DG-1350 recommends that an applicant using a PRA to define the spectrum of credible accidents justify that the PRA is acceptable for this use, and consider internal and external hazards, all modes of operation, and all significant radionuclide sources.

(c) It is recognized that a PRA for a nuclear plant in the design-phase of development (e.g., design certification application, standard design approval application, combined license application) may not satisfy each technical requirement of the ASME/ANS PRA standard to meet current good practice (i.e., Capability Category II). However, as outlined in RG 1.200, for some applications Capability Category I may be acceptable for some requirements. As described in Section 2.5 of the LTR, there are two obvious areas where a new nuclear plant design will not be able to meet Capability Category II of the ASME standard - operational experience and plant walkdowns. As the LTR requires, applicants need to demonstrate that the PRA is sufficient to support the risk-informed decision making in the EPZ methodology.

In the case where there is no plant-specific operating experience or data, supporting requirements (SRs) can be met using generic and/or similar plant data/operating experience with associated uncertainties and assumptions. Similarly, where walkdowns cannot confirm aspects of the analysis, SRs can be met using generic information and general design documents with associated uncertainties and assumptions. Analyses should consider information relating to spatial aspects of structures, systems, and components if that information is available. This approach meets the objective of these supporting requirements for a plant in the preoperational phase. In addition, screening is only applicable to PRA accident sequences; the EPZ methodology also requires evaluation of the design basis source term from Chapter 15 of the final safety analysis report, as described in Section 3.3 of the LTR.

Further, as discussed in Section 2.4 and 3.7 of the LTR and the response to question 01.05-45, the EPZ methodology requires a qualitative plant-level evaluation of defense-in-depth to account for PRA uncertainties; the qualitative evaluation of defense-in-depth is performed to confirm that design features and the safety strategy employs successive compensatory measures to prevent accidents or mitigate consequences.

(d) Walkdowns are described in paragraph (c).

[1] Other approaches (i.e., other than RG 1.200) may require an in-depth review by NRC staff to determine PRA technical acceptability.

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, has been revised as described in the response above and as shown in the response to RAI Question 01.05-43.

Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828

Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-47

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening

Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

Both the 2009 version of the Level 1/LERF LWR PRA standard (ASME/ANS RA-Sa-2009) and the current draft of the next edition of the Level 1/LERF LWR PRA standard (ASME/ANS RA-S-1.1) include the terms significant accident sequence, significant accident progression sequence, significant cut set and the definitions thereof. The definitions of those terms have built into them quantitative criteria related to when something is considered to be significant and, therefore, need to be considered (i.e., cannot be dismissed). Specifically, the criteria classify a contributor as significant when they are part of the top 95 percent of the total risk or any individual contributor is more than 1 percent of the total risk. According to the LTR, Section 3.4.3, "Screening of Single Module Accident Sequences on Core Damage Frequency," all significant internal and external accident sequences can be screened if they have a core damage frequency less than 1E-7 per year. The staff is concerned that parsing of core damage sequences into individual components for comparison against numerical screening thresholds could screen out potentially risk significant core damage sequences.

Request

For ALWRs, the staff is requesting that the LTR be revised to address the concern of parsing sequences to limit parsing and to ensure consistent application of the methodology.

NuScale Response:

The use of individual sequences, rather than sequence families, to support the emergency planning zone (EPZ) methodology is intentional with the goal of simplifying the identification of severe accident and dose consequence simulations required downstream of the screening. If multiple sequences are grouped together for screening, then one must be chosen for severe accident and dose consequence evaluation, creating ambiguity in the EPZ methodology as to which sequence is representative and appropriate for further evaluation.

Sequences used in the screening process are expected to be defined by an initiating event and top events representing the success or failure of mitigating systems at the system level. Any combination of failures of individual components or subcomponents within a system that fail the system mission are considered to belong to the same sequence. Similarly, initiating events within a probabilistic risk assessment (e.g., similar fire events) should be grouped such that all events with similar availability and response of plant systems to an event are grouped. The identification of sequences is clarified in Section 3.4.1 of the EPZ licensing topical report (LTR). Examples of such sequences are provided in Appendix C of the EPZ LTR. During the application of the method, the U.S. Nuclear Regulatory Commission staff reviewers will have the opportunity to review and approve the identification of sequences as appropriate.

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, been revised as described in the response above and as shown in the markup provided in this response.

3.4.1 Compilation of Probabilistic Risk Assessment Accident Sequences

To support the identification of applicable accident sequences, PRA accident sequences for all internal events and external events, over all operating modes, are compiled. The use of a site-specific PRA and all associated accidents is appropriate to identify applicable accident sequences to be considered in the EPZ methodology.

The total CDFs from all initiating events and operating modes are summed to calculate the total CDF per module year (used in the probability of dose exceedance calculation, Section 4.2.3). A point estimate sequence-level CDF, which is calculated to approximate the mean of a distribution, as opposed to a higher percentile value from a PRA uncertainty analysis, is appropriate for a best-estimate EPZ evaluation. The EPZ methodology incorporates a separate uncertainty analysis, which supports the use of a best-estimate evaluation.

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

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

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

Response to Request for Additional Information
Docket: 99902078

RAI No.: 9828
Date of RAI Issue: 04/22/2021

NRC Question No.: 01.05-48

Regulatory Basis:

Title 10 of the Code of Federal Regulations (CFR) 50.47 and 10 CFR Part 50 Appendix E codify the emergency planning requirements for nuclear power reactors. Specifically, the plume exposure emergency planning zone (EPZ) for power reactors generally consists of an area about 10 miles in radius for reactors with an authorized power level greater than 250 megawatts thermal (MWt), and the EPZ may be determined on a case-by-case basis for reactors with an authorized power level less than 250 MWt. If a reactor has an authorized power level greater than 250 MWt, an exemption may be required for an EPZ less than 10 miles in radius. The technical basis for the 10-mile plume exposure EPZ is given in NUREG-0396/EPA 520/1-78-016, "Planning Basis for the Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants" (Agencywide Document Access and Management System (ADAMS) Accession No. ML051390356), and was based upon evaluation of the offsite consequences of accidents (both design basis and severe) and a comparison of doses to the Environmental Protection Agency (EPA) guidance on when to take emergency response actions. The EPA emergency response actions include sheltering and evacuation as given in the Protective Action Guides (PAGs), or, for very low-probability and high-consequence accidents, demonstration that the probability of exceeding a radiation exposure deterministic health effect is low and decreasing at the chosen outer boundary of the plume exposure EPZ. The assumptions and approach used in the analysis of the NuScale EPZ sizing methodology, including the selection of accident sequences for source term calculations, can impact the results.

Discussion

NuScale Power, LLC (NuScale) submitted licensing topical report (LTR) TR-0915-17772-P, Revision 2, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones" for review by the NRC staff. As stated in Section 3.0, "Accident Screening

Methodology," of the LTR, a combined license (COL) applicant can use this risk-informed approach to select appropriate single and multi-module accident sequences for the EPZ technical basis based on their Probabilistic Risk Assessment (PRA) which is expected to cover all internal and external initiators. The Executive Summary further states that, "The methodology is intended for use by all advanced nuclear reactor designs, particularly small modular reactors (SMRs) such as NuScale, although it is acknowledged that not every element of the method will be applicable to all designs."

Issue

The staff did not find information in the LTR about potential releases due to non-core damage events that would necessitate protective actions.

Request

The staff is requesting that the LTR include potential releases due to non-core damage events that would necessitate protective actions consistent with the Environmental Protection Agency (EPA) Protective Action Guidance (PAGs). For example, dropping the upper portions of the NuScale reactor pressure vessel and the containment vessel as they are moved to or from the dry dock area, onto the fuel in the lower Reactor Pressure Vessel (RPV), which remains in the refueling flange tool may cause mechanical fuel damage and a gap release.

NuScale Response:

The emergency planning zone (EPZ) methodology includes the evaluation of a representative design-basis release discussed as in Section 3.3 of the EPZ licensing topical report (LTR) (representing both core damage [CD] releases and non-CD releases) as at least one of the source terms for determining the size of the EPZ. Any other potential releases (either CD or non-CD) are then (by definition) a result of a beyond-design-basis event (BDBE). The PRA evaluates BDBEs that are CD events and considers them in the EPZ methodology as discussed in Section 3.4 of the EPZ LTR. The BDBEs that are non-CD events and outside the PRA are considered other risks, and would be evaluated as discussed in Section 3.5 of the LTR; Section 3.5 has been expanded to clarify other risks must be identified and evaluated to ensure an appropriate planning distance. Finally, BDBE releases (either CD or non-CD) would either be shown to be bounded or evaluated against the appropriate dose criterion (e.g., a bypassed containment would be considered "more severe" and evaluated against dose criterion c in Section 3.2 of the LTR.)

Impact on Topical Report:

Topical Report TR-0915-17772, Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites, been revised as described in the response above and as shown in the markup provided in this response.

3.5 Other Risks

Other risks with impact to EPZ sizing outside of the events identified in Section 3.3 and Section 3.4 may exist that are design or site-specific. These other risks that may lead to potential off-site radionuclide releases must be identified and evaluated to ensure an appropriate planning distance. Applicants demonstrate acceptable consideration of such risks by meeting one of the following criteria:

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

This report provides acceptance criteria for two such risks, using the second criterion above, that will be common for some advanced reactor designs: SFPs, and severe accident phenomena. Applicants with other risks in addition to the SFPs and severe accident phenomena demonstrated in the following sections shall~~will need to~~ propose acceptance criteria and demonstrate~~demonstration of~~ acceptability for those risks.

3.5.1 Spent Fuel Pool

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

- The time required to boil off SFP inventory is sufficiently long (i.e. >10 hours) that mitigating measures can be implemented to prevent fuel damage.
- Criticality is precluded during refueling operations and storage of spent fuel at maximum capacity, including following design-basis events.
- Leaks from the SFP can be detected and mitigated through monitoring and replenishment systems.

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

3.5.1.1 Spent Fuel Pool Boil-Off

Under this scenario, all 12 reactor modules are simultaneously shut down with no active cooling of the ultimate heat sink (UHS). The decay heat from the reactor modules and the maximum capacity of the SFP are the heat source. The UHS heats up and eventually evaporates with the conservative assumption that no water returns to the UHS from condensation in the 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.

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

Enclosure 3:

Affidavit of Mark W. Shaver, AF-119894

NuScale Power, LLC

AFFIDAVIT of Mark Shaver

I, Mark Shaver, state as follows:

1. I am the Manager, Licensing 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 Request for Additional Information response reveals distinguishing aspects about the by which NuScale develops its Response to EPZ Topical Report RAIs.

    NuScale has performed significant research and evaluation to develop a basis for this 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 response to NRC Request for Additional Information eRAI-9828 EPZ. 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 June 10, 2022.

Mark W. Shaver

Mark Shaver