~~PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390~~

December 22, 2015 Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-0915-17772, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," Revision 0 (NRC Project No. 0769)

REFERENCE: Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Key Issue Resolution Prior to Design Certification Application," LO-0715-16060, dated July 22, 2015 (ML15203B306).

In Attachment 2 of the referenced letter, NuScale Power, LLC (NuScale) provided an updated schedule for topical report submittals. Consistent with that schedule, NuScale indicated its intent to submit Topical Report TR-0915-17772, "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," Revision 0 by December 31, 2015. In light of its early submittal and the holiday season, NuScale respectfully requests that the acceptance review be completed in about 60 days from the beginning of the new calendar year 2016.

The purpose of this submittal is to request the Nuclear Regulatory Commission (NRC) review and approval of the NuScale design-specific, plume exposure Emergency Planning Zone (EPZ) sizing methodology which will be used by COL applicants. While not required for the Design Certification Application (DCA), review in parallel with the DCA will be efficient and best supports Combined License (COL) applicant needs. NuScale also requests, as part of this review, that the NRC provide a safety evaluation report (SER) on the design-specific, plume exposure EPZ sizing methodology, including the following:

1. A conclusion that the NuScale proposed plume exposure EPZ methodology in the topical report, when supported by design-specific information and appropriately implemented by each COL applicant, is an acceptable approach for justifying the plume exposure EPZ size for the NuScale design.

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

With the EPZ LTR having been approved by the NRC, NuScale and each COL applicant will collaborate to develop emergency preparedness elements with each contributing organization having distinct roles and responsibilities. The COL applicants will apply the approved methodology to their specific sites to determine a plume exposure emergency planning zone.

Enclosure 1 contains the proprietary version of the report entitled "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites." NuScale requests that the proprietary version be withheld from public pursuant to 10 CFR 2.390. The enclosed affidavit (Enclosure 3) supports this request.

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Enclosure 2 is the nonproprietary version of the report entitled "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites."

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

Please feel free to contact Steven Mirsky at 301-770-0472 or at smirsky@nuscalepower.com if you have any questions.

Sincerely,

Thomas A. Bergman
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Enclosure 1: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," (TR-0915-17772-P), Revision 0, proprietary version

Enclosure 2: "Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites," (TR-0915-17772-NP), Revision 0, nonproprietary version

Enclosure 3: Affidavit, AF-1115-19305

Enclosure 1:

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

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Enclosure 2:

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

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Licensing Topical Report

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

December 2015

Revision 0

Docket: PROJ0769

NuScale Nonproprietary

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Department of Energy Acknowledgement and Disclaimer

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Abstract

The purpose of this licensing topical report (LTR) is to provide the technical basis for emergency planning zone (EPZ) sizing methodology for the NuScale small modular reactor (SMR) plant design. Nuclear power plant emergency planning regulatory requirements are codified in 10 CFR Part 50.47 (Reference 7.1.1), and 10 CFR Part 50 Appendix E (Reference 7.1.2). In 10 CFR 50.47(a)(2), the U.S. Nuclear Regulatory Commission's (NRC) determination of acceptability is tied directly to the review and resulting findings by Federal Emergency Management Agency (FEMA). FEMA and the NRC acceptance of the emergency plan (EP) is a prerequisite for approval of a combined operating license (COL) under 10 CFR 52, Subpart C (Reference 7.1.3). Approval of an early site permit under 10 CFR 52, Subpart A (Reference 7.1.4), requires a no-significant impediment for EP assertion. Both 10 CFR 50.47 and 10 CFR 50 Appendix E specify a ten-mile plume exposure EPZ for power reactors, but also provide for a different EPZ size for reactors with a thermal power of 250 MWt or less on a case-by-case basis. The original basis for the ten-mile plume exposure EPZ was developed in NUREG-0396 (Reference 7.1.5), which was released in 1978 and represented a joint collaboration of the NRC and Environmental Protection Agency (EPA). The protective action guideline (PAG) doses of 1 to 5 rem total dose effective equivalent (TEDE) are based on EPA-400-R-92-001 (Reference 7.1.6).

In a December 2013 white paper submittal to the NRC, the Nuclear Energy Institute (NEI) provided a generic methodology for developing a scalable plume exposure pathway EPZ applicable to a light water reactor (LWR), integral pressurized water reactor design (Reference 7.1.7). The white paper was in response to SECY-11-0152 (Reference 7.1.8), in which NRC stated its intent to develop an EP framework for SMRs (LWR only) and that, "the staff believes that it may be appropriate for small modular reactors (SMRs) to develop…reduced EPZ sizes, commensurate with their accident source terms…and dose characteristics." The SECY also stated that it is anticipated that the industry will develop and implement the detailed calculation method for review and approval by the staff. In the Staff Requirements Memorandum (SRM) dated August 4, 2015 (Reference 7.1.9), the Commission approved the staff's recommendation to initiate a rulemaking to revise regulations and guidance for emergency preparedness for SMRs and other new technologies.

This LTR presents the design-specific plume exposure EPZ sizing methodology for the NuScale design. The design-specific methodology is based on the NEI risk-informed EPZ methodology and extends this risk-informed methodology to address the issue of determining the appropriate accident sequences to be evaluated for EPZ in the NuScale design, and to consider a consequence orientation in the approach.[1] This includes: (1) use of quantitative insights on accident sequences from the NuScale design-specific probabilistic risk assessment (PRA); and (2) application of more qualitative, deterministic, engineering insights emphasizing traditional safety margin and layers of defense-in-depth. The quantitative risk results, including accident source terms and off-site dose versus distance, together with deterministic defense-in-depth

[1] SECY-15-0077 (Reference 7.1.10) and SECY-11-0152 (Reference 7.1.8).

evaluation, will serve as the basis for a plume exposure EPZ size that is appropriate for the NuScale SMR design. A nuclear power plant using NuScale's SMR design is comprised of individual NuScale power modules, each producing 50 MWe of electricity (gross) with its own combined containment vessel and reactor vessel, and its own packaged turbine-generator set. The NuScale plant power is of a scalable design where as many as 12 modules can be sequentially added to produce as much as 600 MWe. The LTR methodology is for a 12-module NuScale plant.

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

The results presented are provided to illustrate the methodology. They support NuScale's expectation for the EPZ basis that accidents will be very low in probability and, even if they occur, their consequences will be small based on the slow radioactive releases and their relatively small magnitudes.

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

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

With the EPZ LTR having been approved by the NRC, NuScale and a COL applicant will collaborate to develop emergency preparedness elements with each contributing organization having distinct roles and responsibilities. NuScale expects that using the EPZ sizing methodology, it will support the COL applicant by performing the actual determination of appropriate accidents, the source term and dose calculations, provide the technical justification for a selected EPZ size, and provide design-specific emergency action levels (EAL) and protective action recommendations (PAR). The COL applicant will provide site-specific characterization in support of NuScale's EPZ methodology-based evaluations. Furthermore, NuScale will work with COL applicants in identification and resolution of issues related to the NuScale EPZ technical basis prior to or as part of the COLA proceeding. The COL applicant will work jointly with NuScale in preapplication engagements with the NRC and FEMA.

Executive Summary

The purpose of this licensing topical report (LTR) is to provide a methodology to establish the technical basis for plume exposure emergency planning zone (EPZ) sizing, for the NuScale small modular reactor (SMR) plant design. Nuclear power plant emergency planning regulatory requirements are codified under 10 CFR and in coordination with FEMA. The current regulatory plume exposure EPZ for power reactors is ten 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. As the NuScale small modular reactor is a passive, inherently safe 160 MWt (50 MWe) per module reactor based on defense-in-depth and risk-informed knowledge accumulated in the nuclear industry, NuScale describes a methodology to establish the technical basis for EPZ sizing with calculated results to illustrate its usability.

NuScale requests, as part of the review and associated comment resolution of this licensing topical report, that the NRC provide an SER on the design-specific sizing methodology. NuScale considers that the methodology herein, when supported by design-specific information and appropriately implemented by the COL applicant, is an acceptable approach to EPZ sizing. With the EPZ LTR having been approved by the NRC, NuScale and a COL applicant will collaborate to develop emergency preparedness elements with each contributing organization having distinct roles and responsibilities. NuScale expects that using the EPZ sizing methodology, it will support the COL applicant by performing the actual determination of appropriate accidents, the source term and dose calculations, provide the technical justification for a selected EPZ size, and provide design-specific emergency action levels and protective action recommendations. The COL applicant will provide site-specific characterization in support of NuScale's EPZ methodology-based evaluations. Furthermore, NuScale will work with COL applicants in identification and resolution of issues related to the NuScale EPZ technical basis prior to or as part of the COLA proceeding. The COL applicant will work jointly with NuScale in preapplication engagements with the NRC and FEMA.

The methodology described in this report is based on the 2013 NEI White Paper framework and incorporates concepts from the original, generic 1978 EPZ size basis in that the objective goal is dose-based linked to considerations of consequences. However, the methodology is established per design-specific PRA information supported by a comprehensive evaluation of severe accident scenarios, and related operational response. The methodology also utilizes and meets the objectives for next generation plants as described in INSAG-10.

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

The methodology first determines the appropriate sequences to be evaluated for EPZ in the NuScale design and in summary consists of the following: (1) use of quantitative insights on accident sequences from the NuScale design-specific probabilistic risk assessment (PRA); and (2) application of more qualitative and deterministic engineering insights emphasizing safety margin and levels of defense-in-depth. Thus, both “more probable, less severe” and “less probable, more severe” scenarios are evaluated. The quantitative risk results, which include accident source terms and off-site dose versus distance from the plant, together with the more qualitative and deterministic defense-in-depth evaluations, serve as the basis for a plume exposure EPZ size methodology corresponding to the NuScale SMR design.

The dose criteria for the NuScale EPZ methodology have been defined based on the original EPZ bases, as noted, and the EPA’s Protective Action Guideline (PAG), applied to the scenarios as follows: (1) 1 to 5 rem total effective dose equivalent (TEDE) for a design-basis accident; (2) 1 to 5 rem TEDE for more probable, less severe scenarios; and (3) 200 rem whole body acute dose for less probable, more severe scenarios.

Using the design-specific, risk-informed methodology developed to select appropriate more probable, less severe accidents, NuScale has also developed a method to evaluate the source term and dose consequence. The example results in the appendices indicate very small total core damage (CD) probability. In addition, if accidents were to occur, they would progress slowly such that significant time to invoke emergency operating procedures (EOP) and for the operators to take mitigating actions are available. In fact, the small core inventory of a module, the extended time to CD and effective fission product aerosol removal in a smaller volume, passively-cooled containment vessel contribute to the fact that the doses from these accidents are well below the EPA PAGs.

Further, the LTR defines the methodology for determining the appropriate less probable, more severe accidents to be evaluated for EPZ. Applying the Structured Decision Process described in the methodology, an example assessment of two representative containment bypass sequences in Appendix C indicate that these accidents are not credible (physically plausible) in the NuScale design and thus should not be included in the technical basis for EPZ sizing. This is based on the adequacy of defense-in-depth (substantially independent layers of protection that will prevent and mitigate the accident) {{

}}$^{2(a),(c)}$ While the example assessment indicates these accidents should not be included in the EPZ basis, source term and dose evaluations were performed. This was done not to justify EPZ size, but rather to illustrate the evaluation methodology. Results indicate that if such events reach CD, they progress slowly (time to CD is many hours) and the dose consequence will have significant margin to the 200 rem whole body dose criterion.

Finally, the multi-module accidents are addressed in the NuScale EPZ methodology. The multi-module methodology focuses on multi-module risks associated with common structures and shared systems between modules which are unique to the NuScale design.

In summary, the NuScale methodology for establishing the design-specific technical basis for plume exposure EPZ sizing addresses adequacy of defense-in-depth, uncertainties, and fully considers accident source terms and dose consequences. The methodology, when implemented with design information as part of a COL application, will be complete and sufficient to develop a basis for and to specify the size of the plume exposure EPZ for a NuScale plant.

The methodology developed considers a spectrum of accident sequences that are used for source term evaluation, beginning with selection criteria from design-specific PRA, with due consideration of uncertainties. A deterministic system analysis was also utilized based on scenarios bound by improbable successive passive equipment failures for completeness. The calculated probabilities of accident scenarios considered lead NuScale to conclude that most sequences are not credible because of its defense-in-depth design, and the relatively long time to reach undesirable conditions. This length in time is deemed sufficient for trained operators to take mitigating action, importantly to mitigate its development such as to ensure energy removal from the reactor core, the reactor pressure vessel, and the containment vessel. For the improbable scenarios, deterministic assessment of core damage, and a hypothetical source term was performed. The source term analyses included consideration of the time-scale, its magnitude, and characteristics of releases. Credit was taken for the NuScale plant passive engineered safety features resulting in dose consequences which are well within the PAGs.

The implementation of the NuScale methodology developed for its SMR, as adopted from the NEI generic methodology, indicates that changes to emergency planning requirements are warranted when the technical criteria for emergency planning requirements are modified to account for the lower probability of severe accidents or the longer time period between accident initiation, and smaller release of radioactive material for most severe accidents. The underlying principle of safety-in-design is realized in the NuScale SMR such that application of the methodology establishes a technical basis to justify a reduced plume exposure EPZ that provides the same level of protection to the public against radiological accidents as the currently authorized EPZ for licensed nuclear power plants.

Licensing Topical Report

1.0 Introduction

1.1 Purpose

The purpose of this licensing topical report (LTR) is to provide the sizing methodology and criteria that can be adopted and used by developers and operators of the NuScale small modular reactor (SMR) for establishing the design-specific and site-specific plume exposure emergency planning zones (EPZs). The purpose of submitting this LTR during the preapplication phase is to provide information to the Nuclear Regulatory Commission (NRC) to facilitate efficient and timely review of the NuScale EPZ sizing methodology in conjunction with, yet independent of, the design control document (DCD) as part of the design certification license application. The topical report requests an NRC review of the NuScale design-specific 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.

After the NRC approves this LTR as documented in its SER, it will be suitable for reference by developers and operators of the NuScale SMR to pursue plant operation at sites where EPZs have been established using this sizing methodology. NuScale and a COL applicant will collaborate to develop emergency preparedness elements with each contributing organization having distinct roles and responsibilities. NuScale expects that using the EPZ sizing methodology, it will support the COL applicant by performing the actual determination of appropriate accidents, the source term and dose calculations, provide the technical justification for a selected EPZ size, and provide design-specific emergency action levels and protective action recommendations. The COL applicant will provide site-specific characterization in support of NuScale's EPZ methodology-based evaluations. Furthermore, NuScale will work with COL applicants in identification and resolution of issues related to the NuScale EPZ technical basis prior to or as part of the COLA proceeding. The COL applicant will work jointly with NuScale in preapplication engagements with the NRC and FEMA.

1.2 Scope

This report provides a design-specific methodology for determining an appropriate plume exposure EPZ for a NuScale plant. The NuScale methodology is based on the application of the NEI risk-informed EPZ methodology (Reference 7.1.7).

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

- methodology, based on the NEI white paper, designed to be structured and repeatable
- NRC EPZ documents (NUREG-0396, SECY-97-020 (Reference 7.1.11), SECY-11-0152, and SECY-15-0077
- risk-informed methods to determine appropriate accidents to be evaluated
- multi-module events and external events
- operationally-focused accident mitigation capability (defense-in-depth)

- analysis of uncertainties

The main body of the LTR contains the design-specific EPZ size methodology for which NRC approval is sought. To illustrate how the EPZ size methodology would be used by future applicants, example source term and dose evaluations and example assessments of appropriate accidents to be evaluated are included in Appendices A, B, and C. The information in the appendices is provided to facilitate: (1) NRC's review of the design-specific EPZ size methodology in the main body for which approval is sought; and (2) an understanding of how this LTR would be implemented by future applicants. NuScale is not seeking NRC approval of the information in the appendices, as the final design information will be presented in the NuScale DCA, and the request for approval of EPZ size will be part of the COL application. The LTR is not part of the design certification but uses information with which it is consistent. This report does not address the ingestion exposure pathway EPZ.

1.3 Abbreviations and Definitions

Table 1-1. Abbreviations

Term	Definition
AIA	aircraft impact assessment
ALWR	advanced light water reactor
AOO	anticipated operational occurrence
AOP	abnormal operating procedure
ARP	alarm response procedure
ASME	American Society of Mechanical Engineers
ATWS	anticipated transients without scram
BDBEE	beyond-design-basis external event
BDG	backup diesel generator
CCDP	conditional core damage probability
CCF	common-cause failures
CCFP	conditional containment failure probability
CD	core damage
CDF	core damage frequency
CDP	core damage probability
CES	containment evacuation system
CFD	containment flooding and drain
CFDS	containment flooding and drain system
CIV	containment isolation valve
CLRF	conditional large release frequency
CNV	containment vessel
COL	combined operating license
CR	control room

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Term	Definition
CSDRS	certified seismic design response spectra
CTG	combustion turbine generator
CVCS	chemical and volume control system
DBA	design-basis accident
DBT	design-basis threat
DC	design certification
DCA	design certification application
DCD	design control document
DHRS	decay heat removal system
EAD	exclusion area distance
EAL	emergency action level
ECCS	emergency core cooling system
EDMG	extensive damage mitigating guideline
ELAP	extended loss of AC power
EOP	emergency operating procedure
EP	emergency planning
EPA	Environmental Protection Agency
EPZ	emergency planning zone
ERO	emergency response organization
ESBWR	economic simplified boiling water reactor
FCI	fuel coolant interaction
FEMA	Federal Emergency Management Agency
FOAKE	first-of-a-kind engineering
FW	feedwater
FWIV	feedwater isolation valve
GMRS	ground motion response spectrum
HCLPF	high confidence of low probability of failure
HSI	human-system interface
IAEA	International Atomic Energy Agency
INSAG	International Nuclear Safety Advisory Group
ISI	in-service inspection
ISRS	in-structure response spectra
LERF	large early release frequency
LHS	Latin hypercube sampling
LOCA	loss-of-coolant accident
LPMS	less probable, more severe
LPSD	low power shutdown
LPZ	low population zone
LTR	licensing topical report
LWR	light water reactor

Term	Definition
MARS	multivariate adaptive regression splines
MCC	motor control center
MGL	Multiple Greek Letter
MM	multi-module
MPLS	more probable, less severe
MS	main stream
MSIV	main steam isolation valve
NDE	nondestructive examination
NEI	Nuclear Energy Institute
NOP	normal operating procedure
NPM	NuScale Power Module
NPP	nuclear power plant
NRC	Nuclear Regulatory Commission
NuScale	NuScale Power, LLC
ORO	off-site response organization
PAG	protective action guides
PAR	protective action recommendation
PCT	peak cladding temperature
PDF	probability density function
PGA	peak ground acceleration
POS	plant operating states
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
RBC	reactor building crane
RCS	reactor coolant system
RF	response factor
RG	regulatory guide
RPS	reactor protection system
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RSV	reactor safety valve
RVV	reactor vent valve
RXB	reactor building
SAMG	severe accident management guideline
SAR	safety analysis report
SBO	station blackout
SCC	seismic correlation class
SCDF	seismic core damage frequency
SER	safety evaluation report
SFP	spent fuel pool

Term	Definition
SG	steam generator
SGTF	steam generator tube failure
SMA	seismic margin assessment
SMR	small modular reactor
SOARCA	state-of-the-art reactor consequence analyses
SPRA	seismic probabilistic risk assessment
SRM	Staff Requirements Memorandum
SRS	simple random sampling
SRV	safety relief valve
SSC	structures, systems, and components
SSE	safe-shutdown earthquake
TAF	top of active fuel
TCD	time (to) core damage
TEDE	total effective dose equivalent
UHS	ultimate heat sink
USGS	U.S. Geological Survey
WB	whole-body

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.
anticipated operational occurrences	Conditions of normal operation that are expected to occur one or more times during the life of the nuclear power unit.
baseline PRA	For a nuclear power plant at the design certification (DC) or combined operating license (COL) stage, where the plant is not built or operated, the baseline PRA model reflects the as-designed plant.
basic event	An element of the PRA model for which no further decomposition is performed, because it is at the limit of resolution consistent with available information. There are typically two types of basic events: equipment unavailability; and human errors.
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, and large release given the knowledge that a prior event has occurred.
core damage	Any physical disruption of the nuclear core including fuel, fission

Term	Definition
	products, and containing geometry consisting of cladding, structure, and flow alignment that could release significant amounts of radioactivity from fission products by being undercooled or overreactive.
core damage frequency	The sum of the accident sequence frequencies of those accident sequences whose end state is core damage.
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 (SSCs) perform as required.
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 100.11 (Reference 7.1.12) excluding reactor coolant pressure boundary and reactor protection system items.
event sequence	A specific event tree pathway in a PRA model that begins with an initiating event in a specified plant operating state and describes the successful and unsuccessful responses of the SSCs that perform safety functions in response to the initiating event and ends in a well-defined end state.
external hazard	An event or a natural phenomenon that poses some risk to a facility. External hazards include events such as flooding and fires external to the plant, tornadoes, earthquakes, and aircraft crashes.
FLEX	An approach for adding diverse and flexible mitigation strategies for mitigating and coping with beyond-design-basis events.
high confidence of low probability of failure	A measure of seismic capacity of a structure, system, or component, expressed in terms of a threshold earthquake intensity, below which failure of the structure, system, or component is highly unlikely.
internal hazard	An event that poses some risk to a facility. Internal hazards include events such as equipment failures, human failures, and flooding and fires internal to the plant.
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

Term	Definition
	radiological doses to the public.
plume exposure pathway EPZ	For nuclear power reactors the plume exposure pathway EPZ is an area of about ten 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 frequency of 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 health and safety.
safe-shutdown earthquake	The maximum earthquake for which certain structures, systems, and components are designed to remain functional.
Seismic Category I	Structures, systems, and components that are designed to remain functional if an SSE occurs.
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 in which substantial damage is done to the reactor core regardless of whether the event results in serious off-site radiological consequences.

2.0 Background

Emergency planning for protective actions within zones around a nuclear power plant has been an NRC requirement since the early 1960s. Initially, 10 CFR Part 100 (Reference 7.2.1) required that every site must have an exclusion area and a low population zone (LPZ).

NUREG-0396 (Reference 7.1.5), 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 7.2.2), recommended a plume exposure pathway EPZ of about ten miles (16 kilometers) and an ingestion exposure pathway EPZ of about 50 miles (80 kilometers). The EPZs were established at ten and 50 miles to provide dose savings to the population in areas where the projected dose from design-basis accidents could be expected to exceed the applicable protective action guides (PAG) (1 and 5 rem total effective dose equivalent (TEDE) under unfavorable atmospheric conditions. NUREG-0396 states that, “As in the DBA-LOCA (design-basis accident loss-of-coolant accident) class, the doses from ‘melt-through’ releases generally would not exceed even the most restrictive PAG beyond about ten miles from a power plant.”

As stated 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 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 the off-site radiological risk is not new to the NRC. The staff reviewed exemption requests from specific emergency planning requirements from certain reactor licensees that have permanently ceased operations (SECY-14-0066 and SECY-14-0118) (References 7.2.3 and 7.2.4, respectively). The staff reviewed these exemption requests against the requirements in Emergency Plans, 10 CFR 50.47, (Reference 7.1.1); Appendix E, to 10 CFR Part 50; and Emergency Plans, 10 CFR 72.32 (Reference 7.2.5).

Industry believes that siting and building SMRs with appropriate EPZ size and planning elements will have benefits for all stakeholders. This is based on the expectation that the SMR overall safety case and defense-in-depth, including design, operation, security, and appropriate EPZ and planning elements, will further enhance the design and safety margins and further reduce accident risk to the public. (Reference 7.1.7)

Many of the fundamental factors governing the necessary EPZ footprint are similar among the current fleet of licensed nuclear power plants; for example, reactor core inventories and reactor containment design parameters. These plants utilize active safety systems, meaning that some motive force is needed to provide reactor and containment cooling, and they have containments designed to withstand internal pressures on the order of 45 psi. In addition, these plants rely to some extent on operator actions to cool the reactor and to prevent containment pressure from reaching design limits. Given their similarities, existing nuclear power plants have similar emergency planning and a ten-mile radius plume exposure EPZ.

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}}[2(a),(c)] In addition, the NuScale design includes fission product barriers not found in conventional pressurized water reactors (PWRs); each containment vessel is submerged in the reactor pool, which is enclosed in the reactor building.

The NuScale design utilizes passive safety features such as natural circulation for core cooling, and does not rely on operator actions to mitigate the effects of a design-basis accident for the first 72 hours following such an event.

The contrasts between the NuScale design and that of operating large nuclear power plants in the fundamental factors that determine EPZ radius, suggest that the EPZ for a NuScale plant can be significantly smaller while providing the same public protection and safety against radiological accidents as the existing nuclear power plant fleet.

The NuScale SMR design offers unique opportunities to optimize emergency planning size and requirements. This optimization will result in:

- smaller plume exposure (and ingestion) EPZ size
- revised emergency action levels (EALs) and technical justification
- revised protective action recommendations (PARs)

- revised emergency response organization (ERO) staffing
- revision of other programmatic aspects of emergency planning (scope or need for 16 planning standards from NUREG-0654, Table B-1 (Reference 7.2.6).

2.1 NuScale Approach

The NuScale design-specific EPZ work was initiated in 2014. The EPZ work is performed in parallel with the DCD preparation, drawing on the DCD design, but not part of design certification. The approach is based on the NEI white paper, NRC feedback, and key NRC EPZ-related documents such as NUREG-0396, SECY-97-020 (Reference 7.1.11), SECY-11-0152 and SECY-15-0077. It incorporates experience and lessons from risk-informed decision-making in regulatory applications, "to determine appropriate accidents to be evaluated" (Reference 7.1.10). It uses a risk-informed evaluation, which balances risk considerations and defense-in-depth by combining insights from:

- assessment of defense-in-depth—deterministic, engineering information emphasizing safety margin
- risk information from the PRA required for new plants

In summary, the NuScale approach relies on:

- NuScale design-specific PRA which considers
  - accident sequences
  - system thermal-hydraulic analyses
- the NEI SMR EPZ sizing methodology
- NuScale design-specific, risk-informed, defense-in-depth evaluation
- identification and evaluations of more probable, less severe scenarios to be addressed for EPZ
- identification of less probable, more severe scenarios and risk-informed assessments of the credibility of these scenarios
- consideration of external events, multi-module accidents, and spent fuel pool (SFP) accidents
- design-specific methodology for operationally-focused mitigation capability
- consideration of uncertainties

2.2 Regulatory Guidance and Requirements

The regulatory framework defining specific EPZ sizes for power reactors was an outcome of a joint task force effort, published in 1978 as NUREG-0396. As stated in SECY-15-0077, in 1979, the NRC issued a policy statement describing two EP planning zones: a plume exposure EPZ of about ten 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, these two EPZs were included in the 1980 rulemaking that added 10 CFR 50.47 to establish standards upon which emergency plans are to be reviewed. Nuclear power plant emergency planning regulatory 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 and resulting findings by FEMA. FEMA and NRC acceptance of the emergency plan (EP) is a prerequisite for approval of a combined operating license (COL) under 10 CFR 52, Subpart C. Approval of an early site permit under 10 CFR 52, Subpart A requires either: a no significant impediments for EP assertion; or a major features EP. Both 10 CFR 50.47 and 10 CFR 50 Appendix E require generally a ten-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.

Sizing requirements for EPZs, stated in 10 CFR 50.47, are based on conservative analyses for large LWRs contained in WASH-1400. Insights from 50-plus years of industry design and operating experience are now available, together with growth of the experimental data based on radionuclide release during an accident and the analytical tools available to calculate such releases in the four decades since the WASH-1400 report was published in 1975.

Further EPZ regulatory guidance is given in NUREG-0396, SECY-97-020, SECY-10-0034 (Reference 7.2.7), SECY-11-0152, and SECY-15-0077. The Planning Basis for The Development of State and Local Government Radiological Emergency Response Plans in Support of Light Water Nuclear Power Plants, NUREG-0396 (Reference 7.1.5), provides a basis for federal, state, and local government emergency preparedness organizations to determine the appropriate degree of emergency response planning efforts in the environs of nuclear power plants.

Since the issuance of NUREG-0396, the staff has conducted several studies useful in evaluating the adequacy of the plume exposure pathway EPZs:

- NUREG/CR-6953 (Reference 7.2.8), Review of NUREG-0654, Supplement 3, Criteria for Protective Action Recommendations for Severe Accidents, Volumes 1, 2, and 3, evaluates the efficacy of various protective action strategies within the EPZ.
- NUREG/CR-6864 (Reference 7.2.9), Identification and Analysis of Factors Affecting Emergency Evacuations, Volumes 1 and 2, examines large evacuations in the U.S. between 1990 and 2003 to gain a fuller understanding of the dynamics involved.
- NUREG/CR-6981 (Reference 7.2.10), Assessment of Emergency Response Planning and Implementation for Large Scale Evacuations, issued March 2008, assessed 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.

- NUREG-1935, State-of-the-Art Reactor Consequence Analysis (SOARCA) Report (Reference 7.2.11), evaluates hypothetical evacuations within EPZs and beyond, in response to a series of accident scenarios. These analyses informed the staff's conclusion that the current requirements for EPZs remain protective of public health and safety.

Results of Evaluation of Emergency Planning for Evolutionary and Advanced Reactors, SECY-97-020, describes the review efforts and includes a detailed discussion of the EPZ sizing rationale first described in NUREG-0396. SECY-97-020 (Reference 7.1.11) also describes the advances in source term and severe accident data and summarizes industry submittals.

Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs, SECY-10-0034, informs the Commission of potential policy, licensing, and key technical issues that may require Commission consideration to support future design and license review applications for SMRs, and the staff's plans for developing plans for their resolution.

Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors, SECY-11-0152, discusses the NRC staff's intent to develop a technology-neutral, dose-based, consequence-oriented EP framework for SMRs.

Options for Emergency Preparedness for Small Modular Reactors and Other New Technologies, SECY-15-0077, proposes a consequence-based approach to establishing requirements, as necessary, for off-site EP. SECY-15-0077 requests that the Commission authorize a rulemaking effort to establish EP requirements for SMRs and other new technologies that are commensurate with the potential consequences to public health and safety, and the common defense and security at these facilities. In the Staff Requirements Memorandum (SRM) dated August 24, 2015 (Reference 7.1.9), the Commission approved the staff's recommendation to initiate a rulemaking to revise regulations and guidance for emergency preparedness for SMRs and other new technologies.

2.3 NEI White Paper

The white paper describes a proposed methodology and criteria for establishing the technical basis associated with small modular reactor (SMR) emergency planning zone (EPZ) sizing. The white paper is in support of the continuing dialogue with the NRC on emergency preparedness and SMR-appropriate EPZs, and responds to the Development of an Emergency Planning and Preparedness Framework for Small Modular Reactors, SECY-11-0152, which discusses the NRC staff's intent to develop an EP framework for SMRs. The paper addresses SMRs with light-water-cooled and moderated designs only, and is not applicable to other types of SMRs, nor to large light water reactors (LLWRs). The technical basis for determining the EPZ size which 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 and smaller, slower, fission product releases in 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 probabilistic risk assessment (PRA) required for new plant designs 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.

2.4 Recent NRC SMR EPZ-Related Documents

The most recent NRC SMR-related document is SECY-15-0077. The purpose of this paper was to seek the NRC Commission approval of the staff's recommendation to initiate a rulemaking to revise regulations and guidance for emergency preparedness for small modular reactors (SMRs) and other new technologies, such as non-light water reactors and medical isotope production facilities.

The staff proposes revising NRC regulations and guidance through rulemaking to require SMR license applicants to demonstrate how their proposed facilities achieve EPA PAG dose limits at specified EPZ distances, which may include the site boundary. This framework 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 this EP framework would be developed as part of rulemaking. This would include quantitative guidelines and criteria for accident selection and evaluation specific to SMRs and other new technologies. The NRC technical staff will review design and licensing information to ensure that the information applicants provide on the off-site dose consequences is commensurate with the requested EPZ size and that the applicable requirements ensure adequate protection of public health and safety, and the environment. The Commission's direction regarding EP for SMRs and other new technologies, including EPZ sizes, will enable the NRC staff to develop regulations and guidance in the licensing process. In the Staff Requirements Memorandum (SRM) dated August 24, 2015, the Commission approved the staff's recommendation to initiate a rulemaking to revise regulations and guidance for emergency preparedness for SMRs and other new technologies.

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5.0 Design-Specific Methodology for Operationally-Focused Mitigation Capability

5.1 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. 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 the current fleet guidance that exists in emergency operating procedures, severe accident management guidelines, extensive damage mitigation guidelines and FLEX support guidelines. For clarity, the current terminology of this guidance will be referred to in this report.

Plant design

The NuScale plant is an innovative design based on 50 years of practical application of light-water-cooled pressurized-water reactor (PWR) technology. The design incorporates several features that reduce complexity, improve safety, and enhance operability. The NuScale design philosophy includes:

- a design using proven standard technology.
- smaller reactor core size.
- a below-grade containment immersed in an ultimate heat sink pool of water.
- passive safety systems.
- no operator action required for at least 72 hours following a postulated accident.
- 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.

Design-basis accident mitigation

The NuScale SMR design relies on 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 three primary systems that, by design, mitigate accidents are: decay heat removal; emergency core cooling; 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 events analyzed in Chapter 15 are successfully mitigated without operator intervention required.

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5.2 Training, Equipment, and Procedures

Operators' ability to take appropriate actions to effectively complement the functions of the passive systems to mitigate or prevent undesirable consequences of severe accidents will depend largely on 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.

5.2.1 Operator Training, Staffing, and Qualification

Staffing and qualification:

- The on-site operations minimum staffing assessment for the 12-module NuScale design will include licensed and nonlicensed operators as follows:
  - one shift manager (SRO license)
  - one control room supervisor (SRO license)
  - one shift technical advisor (SRO license and technical degree)
  - three reactor operators (RO license)
  - four nonlicensed operators

Training and qualification of operations personnel will be an Institute of Nuclear Power Operations (INPO) accredited program in accordance with:

- The Process for Initial Accreditation of Training in the Nuclear Power Industry, ACAD 08-001 (Reference 7.5.1)
- The Process for Accreditation of Training in the Nuclear Power Industry, ACAD 02-002 (Reference 7.5.2)
- Guidelines for Initial Training and Qualification of Licensed Operators, ACAD 10-001 (Reference 7.5.3)
- Guidelines for Continuing Training of Licensed Personnel, ACAD 07-001 (Reference 7.5.4)

In addition, shift manager selection and training will be per Guidelines for Shift Manager Selection, Training and Qualification, and Professional Development, ACAD 97-004 (Reference 7.5.5); shift technical advisor training will be per Guidelines for the Training and Qualification of Shift Technical Advisors, ACAD 14-002 (Reference 7.5.6); and nonlicensed operator training will be per Guidelines for Training and Qualification of Nonlicensed Operators, ACAD 15-009 (Reference 7.5.7).

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. Training would also include the basis of the emergency planning zone sizing.

5.2.2 Plant Monitoring Systems

The postaccident monitoring variables are developed in accordance with Regulating Guide 1.97, Revision 4 (Reference 7.5.8). This revision of the regulating guide selects postaccident monitoring variables on a performance basis. The selection of the NuScale SMR postaccident monitoring variables is done in conjunction with the Operations group. The Operations group is developing the procedures for accident mitigation. This is done to ensure that the list of variables developed will allow proper implementation of the accident mitigation procedures. (Refer to Table 5-1 for candidate postaccident monitoring variables.)

5.2.3 Procedures

The following is brief description of the types of procedures that will be available to the 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 warning. ARPs direct actions to take in response to a particular alarm or warning. 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 design-basis accidents, beyond-design-basis accidents 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 following flow diagram depicts the structure of the procedures.

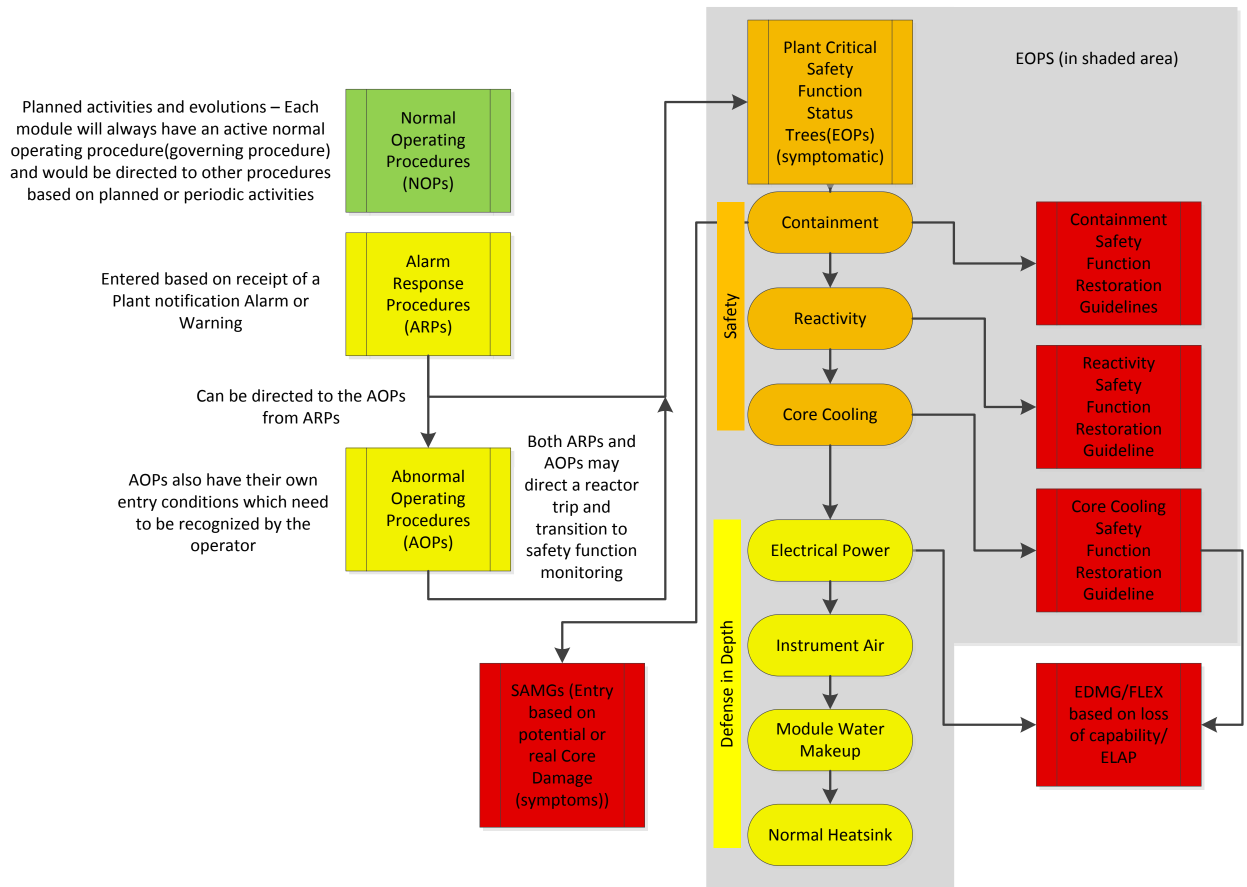

Figure 5-1. Procedure implementation flow diagram

The following three procedure sets will be monitored and entered based on symptoms—these symptoms will be based on plant parameters for the emergency operating procedure (EOPs) and severe accident management guidelines (SAMGs), or based on loss plant capabilities for the FLEX support guidelines. The procedures are broken out 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 will be primarily made up of the operator actions assumed in the probabilistic risk assessment (PRA). The three critical safety functions (in Table 5-1 below) monitored in the EOPs are core cooling, containment integrity, and reactivity control.

Critical safety function monitoring will be initiated any time a safety system actuation has occurred or conditions indicate that one is required. Actions taken to recover a function are primarily the actions modelled in the PRA fault trees, these actions will be categorized as follows:

Table 5-1. {{

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Monitoring will 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 or it can be used to restore defense-in-depth by restoring normal heat sink and placing DHR back in standby. The NuScale design has several AC power sources available, two backup diesel generators, which can power certain important-to-safety loads, an alternate AC power source which can restore power to any AC electrical bus, and is sized to restart a unit. The operator may also elect to restart a unit and use it to supply AC power to the site.

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. In the current light water reactors, these procedures shift operator focus to preservation and enhancement of containment of the damaged core. Examples considered in the current fleet include: flooding containment to improve in-vessel core retention; venting containment to prevent rupture; and, attempts to inert containment to prevent hydrogen deflagration. 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 being considered is the addition of a reactor building 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—the operator takes action to restore the capability. FLEX support guidelines will be developed during the COLA. They will support activities as described in Chapter 20 of the NuScale DCD.

NuScale design features support operator mitigation strategies in the performance of the following FLEX actions:

- ultimate heat sink makeup
  - connect a portable pump to provide makeup to the reactor building pool
  - connect a portable generator to the 480V backup diesel generator motor control centers
- additional capabilities recognized and under consideration
  - RCS inventory makeup and reactivity control
  - makeup to the RCS with external power
  - flood containment with external power

The FLEX strategy is discussed in Section 3.8.7.

5.2.4 Examples of Scenarios where Operator Actions are Possible

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5.2.5 Operating Experience

This section addresses the actions NuScale has taken to address the lack of plant specific operating experience as it relates to the development of the EPZ.

The smaller source term and long 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 anticipated to affect multiple modules. Nonsafety-related systems that are needed for electrical production and the ultimate heat sink may affect multiple modules. These shared systems either do not contribute to or are not credible in resulting in core damage. No operator action is credited in any of the design-basis analysis results to mitigate core damage.

Industry Operating Experience

The human factors engineering (HFE) program that supports designing and inspecting the main control room operator interfaces utilizes an extensive operating experience review (Reference 7.5.9). 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 modelling is available.

NuScale Operating Experience

NuScale is using the many years of combined experience of previous license holders as input into the design of the controls, procedures, emergency plan, and conduct of operations. This experience is used to influence the operator interface to allow for quick diagnosis and communication of accident conditions.

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 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 which aid the operator to perform actions that are correct, including:

- alarm logic that only annunciates when action is required
- procedures integrated into the control interface
- emergency operating procedures that are symptom based
- only three safety functions required to monitor for core damage
- all DBA scenarios require no operator action to prevent core damage

Emergency Action Levels (EALs)

Federal regulations require that a nuclear power plant operator develop a scheme for the classification of emergency events and conditions (Reference 7.5.9). The NuScale EAL scheme is based on two schemes developed by the Nuclear Energy Institute and endorsed by the NRC (Reference 7.5.10 and 7.5.11). Neither of these schemes is applicable to small modular reactors due to significant design and operating differences. However, the NuScale scheme relies on the developmental guidance these documents offer and is modeled after them as closely as possible. The development process produced initial EALs that are simpler to use with fewer accident sequences that can lead to core damage than the schemes used at conventional facilities. The NuScale design-specific EAL scheme will be provided to NEI as part of an ongoing project to develop new NEI guidance for SMR EALs.

5.2.6 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, or operator actions. Multiple passive safety systems must fail in order for damage to occur. Even in the extremely unlikely event of core damage, the time to start of fission product release is many hours, which is significantly 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. The number and type of actions that the operators can take are fewer and simpler than at current large light water reactors by virtue of the fact that there are fewer systems and components in the NuScale design.

NuScale has incorporated best practices obtained from operational experience both external and internal, and applied as appropriate throughout all portions of the design. This has led to simple safety systems and intuitive controls in both the physical plant and operator interface. The EPZ size is determined based on a function of source term and time from core damage to radioactive release. A NuScale facility has a smaller source term and much longer accident progression sequences when compared to current nuclear electricity generating facilities. This supports a technical basis for justifying a reduction in the current EPZ size to one more appropriate for a NuScale facility that provides at least the same degree of public safety as the current EPZ size does for large light water reactors.

6.0 Summary and Conclusions on Methodology

The NuScale proposed approach for developing the technical basis for EPZ size utilizes the 2013 NEI white paper framework and incorporates concepts from the original, generic 1978 EPZ size basis in that it is dose-based and has a consequence orientation. At the same time, important differences exist in the NuScale approach including that: it is design-specific, utilizing design-specific PRA information; it applies the severe accident knowledge base and analytical methods developed over the four decades since the original EPZ basis was formulated; and it is designed to be comprehensive and repeatable. In addition, given the extent of PRA development and the evolution of risk-informed regulatory applications over the last several decades, NuScale is using risk-informed methods for determining appropriate accidents to be evaluated for the EPZ size basis. This risk-informed approach includes PRA information and a deterministic assessment of the adequacy of defense-in-depth in preventing and mitigating less probable, more severe accidents that are addressed in the EPZ sizing methodology.

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

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

Overall, the design-specific EPZ methodology, as proposed in this LTR and to be implemented with detailed design information as part of a COL application, is a complete and sufficient approach for developing the basis for and specifying the size of the EPZ for a NuScale plant. In addition, based on representative source term and dose evaluations, accidents determined to be appropriate for evaluation as part of the EPZ size basis are very low in probability and even if an accident occurs, the releases will be small and slow, with CD not occurring for many hours.

The following conclusions can be drawn regarding the NuScale proposed approach for developing the technical basis for 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 6-1.

Table 6-1. Summary of dose criteria for NuScale EPZ methodology

Accident Type	Dose Criteria
DBA	1 to 5 Rem TEDE
More Probable, Less Severe	1 to 5 Rem TEDE
Less Probable, More Severe	200 Rem Whole Body Acute

2. A design-specific, risk-informed methodology for determining more probable, less severe accidents to be evaluated has been defined, along with the methodology for source term and dose evaluation. Example results in Appendix A indicate a very low total CD accident probability and that accidents, if they do occur, progress slowly (many hours to start of CD), thus providing significant time for EOP and operator mitigation actions.
3. Given the small core inventory of a module, the extended time to CD, and the effectiveness of fission product aerosol removal in the smaller volume, passively cooled containment, the example results indicate that doses from these accidents will be well below the PAGs and significantly less than doses for the large operating plants. This is supportive not only of the EPZ size basis, but also the INSAG-10 objective for next generation plants that only protective measures that are very limited in scope in terms of both area and time would be needed.
4. A design-specific, risk-informed methodology for determining less probable, more severe accidents to be evaluated has been defined. Based on application of the Structured Decision Process presented in Figure 3-2, example assessments of two containment bypass sequences in Appendix C indicate that these accidents are not credible (physically plausible) in the NuScale design. Thus, source term and dose evaluations for these accidents would not be performed and the accidents would not be part of the EPZ size basis.
5. While the Appendix C assessment of the two containment bypass accidents indicates that these accidents should not be included in the EPZ basis, preliminary source term and dose evaluations were performed on these accidents. This was done not to justify EPZ size, but rather to illustrate the evaluation methodology. These evaluations are described in Appendix B. Results indicate that if such accidents reach CD, they progress slowly (time to CD is many hours) and that the dose will have significant margin to the 200 rem whole body dose criterion.
6. Application of INSAG-10 is specified for further improvement in defense-in-depth for NuScale plants.
7. The methodology specifies a single-module seismic probabilistic risk analysis that includes seismic hazard profile, seismic fragility analysis, seismic systems analysis, and seismic risk quantification.
8. A methodology for addressing multi-module risk has been developed which focuses on multi-module risks associated with common structures and shared systems between modules which are unique to the NuScale design. The methodology includes a bounding multi-module core damage scenario for EPZ purposes involving

three modules becoming damaged due to a dropped module accident during refueling. The module dropped by the crane is assumed to be damaged as well as two operating modules in opposing bays in the reactor pool.

9. Other PRA risks were identified and analyzed. The approach proposed to analyze these risks includes the methodology for ATWS events, external natural hazards such as tornado and hurricane, spent fuel pool, low power and shutdown (LPSD), steam explosion, hydrogen in reactor building, and correlated hazards.
10. Risks outside the PRA are addressed including security events and aircraft impact.
11. Design-specific methodology for source term and dose evaluations that include software qualification and use of codes (such as NRELAP5, MELCOR, and MACCS2) for the events identified, including an uncertainty analysis methodology.
12. A design-specific methodology for operationally-focused mitigation of severe accidents is specified that is based on deterministic rather than probabilistic considerations.

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 assessments of less probable, more severe accident credibility are provided in Appendix C. It should be noted that the results presented in Appendix A, Appendix B, and Appendix C are solely intended to demonstrate the methodology and are not intended as the basis for a NuScale design-specific plume exposure EPZ size.

7.0 References

Referenced documents are given in the order of appearance.

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There are no references in Section 6.0.

7.7 Appendices Document References

Appendix A Document References

There are no references in Appendix A.

Appendix B Document References

There are no references in Appendix B.

Appendix C Document References

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Appendix A. {{

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Enclosure 3:

Affidavit, AF-1115-19305

NuScale Power, LLC

AFFIDAVIT of Thomas A. Bergman

I, Thomas A. Bergman , state as follows:

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

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

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

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

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

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

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

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale’s competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the NuScale’s plant design and engineering analyses which were used to illustrate the methodology for establishing the technical basis for plume exposure emergency planning zones at plant sites where its small modular reactors may be located.

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

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

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

(4) The information sought to be withheld is in the enclosed report entitled “Methodology for Establishing the Technical Basis for Plume Exposure Emergency Planning Zones at NuScale Small Modular Reactor Plant Sites.” The enclosure contains the designation “Proprietary" at the

top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

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

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

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

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

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

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

(e) Public disclosure of the information is likely to cause substantial harm to the competitive position of NuScale, taking into account the value of the information to NuScale, the amount of effort and money expended by NuScale in developing the information, and the difficulty others would have in acquiring or duplicating the information. The information sought to be withheld is part of NuScale's technology that provides NuScale with a competitive advantage over other firms in the industry. NuScale has invested significant human and financial capital in developing this technology and NuScale believes it would 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 December 22, 2015.

Thomas A. Bergman