LO-0416-48544

~~PROPRIETARY INFORMATION – WITHHOLD UNDER 10 CFR 2.390~~
~~EXPORT CONTROLLED INFORMATION – WITHHOLD UNDER 10 CFR PART 810~~

April 8, 2016

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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11555 Rockville Pike
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SUBJECT: NuScale Power, LLC Submittal of Topical Report TR-09-15-17565, "Accident Source Term Methodology," Revision 1 (NRC Project No. 0769)

REFERENCE:

1. Letter from NuScale Power, LLC to U.S. Nuclear Regulatory Commission, "Request for Suspension of Acceptance Review of TR-0915-17565-P, "Accident Source Term Methodology," Revision 0" LO-0316-483177, dated March 24, 2016 (ML16084B005).

2. NuScale Topical Report "Accident Source Term Methodology," TR-0915-17565-P, Revision 0 (ML16004A218).

In Reference 1, NuScale Power, LLC (NuScale) requested suspension of the acceptance review of topical report TR-0915-17565-P, "Accident Source Term Methodology," Revision 0 (Reference 2) and communicated the intent to submit Revision 1 of the topical report by April 8, 2016.

The purpose of this submittal is to request NRC review and approval of the assumptions, codes, and methodologies presented in Revision 1 of the topical report for assessing the source terms and radiological consequences of design basis accidents. Revision 1 of the topical report provides supplemental information to support the NRC's review. NuScale respectfully requests that the NRC resume the acceptance review of this topical report.

Enclosure 1 contains the proprietary version of the report entitled "Accident Source Term Methodology." NuScale requests this enclosure be withheld from public disclosure pursuant to 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 1 has also been determined to contain Export Controlled Information. This information must be protected from disclosure per the requirements of 10 CFR Part 810.

Enclosure 2 is the nonproprietary version of the report entitled "Accident Source Term Methodology."

This letter and its enclosures make no regulatory commitments and no revisions to any existing regulatory commitments.

Please contact Jennie Wike at 541-360-0539 or at jwike@nuscalepower.com if you have any questions.

Sincerely,

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

The Enclosure contains ~~Proprietary Information~~. Upon separation from the Enclosure, this letter is decontrolled.

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Enclosure 1: "Accident Source Term Methodology," TR-0915-17565-P, Revision 1, proprietary version
Enclosure 2: "Accident Source Term Methodology," TR-0915-17565-NP, Revision 1, nonproprietary version
Enclosure 3: Affidavit, AF-0416-48655

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Enclosure 1:

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Enclosure 2:

"Accident Source Term Methodology," TR-0915-17565-NP, Revision 1, nonproprietary version

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Licensing Topical Report

Accident Source Term Methodology

April 2016

Revision 1

Docket: PROJ0769

NuScale Nonproprietary

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This material is based upon work supported by the Department of Energy under Award Number DE-NE0000633.

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Abstract

This NuScale topical report describes the methodology used for establishing the source terms and radiological consequences for a spectrum of design basis accidents. In instances where significant differences between the NuScale small modular reactor design and a large light water reactor cause the methodology to depart from existing regulatory guidance, these departures are justified in detail.

A methodology for establishing the NuScale design basis source term (DBST) release timing and magnitude, which meets the intent of 10 CFR 52.47 (a)(2)(iv), is presented in this report. DBST associated aerosol transport and iodine re-evolution assessment methodologies are also presented. Approval is sought for application of STARNAUA aerosol modeling software for NuScale's range of post-accident containment conditions and the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Approval is also sought for the use of ARCON96 for establishing offsite atmospheric dispersion factors. This topical report is not intended to provide final DBST isotopic inventory values, dose values, atmospheric dispersion factors, or final values of any other associated accident source term evaluation; rather, example values for the various evaluations are provided for illustrative purposes in order to aid the reader's understanding of the context of the application of these methodologies.

Executive Summary

This NuScale topical report describes a generalized methodology for developing accident source terms and performing the corresponding radiological consequence analyses. The methodology is conservative for developing accident source terms. Key unique features of the NuScale methodology are the use of ARCON96 for offsite atmospheric dispersion factors, the development of a design basis source term (DBST) to meet the intent of 10 CFR 52.47 (a)(2)(iv), and the utilization of STARNAUA containment aerosol transport code in the range of NuScale's containment conditions.

For the calculation of offsite atmospheric dispersion factors, current industry practice is to utilize the PAVAN code and methodology that is directly based upon the guidance presented in Regulatory Guide (RG) 1.145 (Reference 7.2.7). PAVAN is conservative, especially at shorter distances, but the large distances typically utilized for offsite radiological consequence analysis have allowed for PAVAN to be a sufficient tool for other applicants. For the calculation of control room atmospheric dispersion factors, current industry practice is to utilize the ARCON96 code and methodology that is based upon the guidance presented in RG 1.194 (Reference 7.2.8). NuScale has smaller offsite distances to consider in offsite radiological consequence calculations than traditional large Light Water Reactors (LWR) and therefore investigated the use of ARCON96 instead of PAVAN to more accurately establish offsite atmospheric dispersion factors. NuScale determined ARCON96 is applicable and conservative for NuScale's intended use of the code. NuScale is seeking NRC approval for this methodology.

10 CFR 52.47 (a)(2)(iv) (Reference 7.2.1) requires nuclear power reactor design certification applicants to evaluate the consequences of a fission product release into the containment assuming the facility is being operated at the maximum licensed power level. 10 CFR 52.47(a)(2)(iv) also requires nuclear power reactor design certification applicants to describe the design features that are intended to mitigate the radiological consequences of an accident. Following the approach of the 2012 NEI position paper on small modular reactor (SMR) source terms (Reference 7.2.20), NuScale utilizes the scenario envisioned in footnote 3 of 10 CFR 52.47 (a)(2)(iv) as the maximum hypothetical accident (MHA). Although the NuScale design may preclude any credible design basis accident scenario that results in substantial core meltdown and fission product release, it is recognized that the analysis of an appropriately determined MHA is necessary to demonstrate that engineered safety features (ESF) provide an acceptable level of protection to the public and control room operators. NuScale plans to utilize the term DBST for the MHA scenario utilized by NuScale to meet the intent of 10 CFR 52.47 (a)(2)(iv).

The MHA has historically been linked to a large break loss-of-coolant accident (LOCA) in large LWRs. However, NuScale has no large diameter primary coolant system piping, therefore a large break LOCA cannot physically be postulated as the basis for this analysis. Rather, NuScale has developed surrogate severe accident scenarios, denoted as source term design basis accidents (STDBA), that meet the regulatory intent to address the MHA as expressed in SRP 15.6.5 and 10 CFR 52.47 for the off-site, control room, and technical support center doses. As discussed in Section 4.2 of this report, the DBST is composed of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of STDBAs that are utilized as inputs in radiological consequence calculations associated with the MHA.

Calculations associated with the DBST take credit for the natural aerosol removal mechanisms inherent in the NuScale containment design. The STARNAUA containment aerosol transport and removal code was benchmarked against experimental data and was shown to be appropriate for modeling aerosol removal in the DBST analysis associated with the post-accident NuScale containment conditions. Consistent with RG 1.183, NuScale utilizes the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon for modeling removal in containment. Through sensitivity analysis on the modeling parameters utilized as input to the STARNAUA code, it was shown that the wide range of valid aerosol modeling parameters utilized were of equal or less importance for the DBST radiological consequence results compared to other key modeling parameters. This insight should reduce the relative importance of the particular aerosol modeling inputs selected for the DBST analysis in the NuScale design certification application.

Example calculations are provided in this report to demonstrate applicability of the methodology and to aid the reader's understanding of the application of these methodologies. The NuScale design certification application referencing this topical report is expected to present design-specific calculations utilizing the methodologies presented herein.

1.0 Introduction

1.1 Purpose

The purpose of this report is to define and justify the methodology for assessing the source terms and radiological consequences of design basis accidents (DBA). NuScale plans to use this methodology in the NuScale design certification application, such as in Chapter 15 of the forthcoming NuScale Design Control Document (DCD). NuScale requests NRC approval that the assumptions, codes, and methodologies presented in this report are technically acceptable and consistent with current regulations.

1.2 Scope

This report describes assumptions, codes, and methodologies utilized to calculate the radiological consequences of design basis accidents. NuScale seeks approval for the methodology for establishing the NuScale DBST release timing and magnitude that meets 10 CFR 52.47 (a)(2)(iv). NuScale also seeks approval of the DBST associated aerosol transport and iodine re-evolution assessment methodologies. Approval is requested for application of STARNAUA aerosol modeling software to NuScale's range of post-accident containment conditions and the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. Approval is also requested for the use of ARCON96 for establishing offsite atmospheric dispersion factors instead of PAVAN.

This topical report is not intended to provide final DBST isotopic inventory values, final dose values, final atmospheric dispersion factors, or final values of any other associated accident evaluation; rather, example values for the various evaluations are provided for illustrative purposes. Radiological consequence dose results and comparisons to 10 CFR 52.47 and General Design Criterion (GDC) 19 limits are provided for illustration to aid the reader's understanding of the context of the application of these methodologies.

A summary of specific positions that NuScale is seeking approval for in this topical report are as follows:

1. {{

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7. {{

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1.3 Abbreviations

Table 1-1. Abbreviations

Term	Definition
ALWR	advanced light water reactor
AST	alternative source term
Bq	Becquerel (unit of radioactivity)
Ci	curie (unit of radioactive decay)
μCi	micro-Curie (1.0E-06 Ci) (unit of radioactive decay)
cfm	cubic feet per minute (unit of flow)
COL	combined license
CR	control room
CVCS	chemical and volume control system
DBA	design basis accident

Term	Definition
DBE	design basis event
DBST	design basis source term
DCA	design certification application
DCD	design control document
DCF	dose conversion factor
DHRS	decay heat removal system
EAB	exclusion area boundary
ECCS	emergency core cooling system
ESP	early site permit
FGR	Federal Guidance Report
FHA	fuel handling accident
GDC	General Design Criterion
HVAC	heating, ventilation and air conditioning
ICRP	International Commission on Radiological Protection
JFD	joint frequency distribution
lbm	pound mass (unit of mass)
LOCA	loss-of-coolant accident
LODC	loss of DC power
LPZ	low population zone
LWR	light water reactor
MHA	maximum hypothetical accident
MSLB	main steam line break
MSIV	main steam isolation valve
MW_{th}	mega-watts thermal (unit of thermal power)
NCDC	National Climatic Data Center
NIST	National Institute of Standards and Technology
NPM	NuScale Power module

Term	Definition
NRC	Nuclear Regulatory Commission (United States)
NWS	National Weather Service
pH_T	concentration of H+ ion on a logarithmic scale (temperature dependent)
PF	partition factor
PRA	probabilistic risk assessment
PWR	pressurized water reactor
REA	rod ejection accident
rem	Roentgen equivalent man (unit of dose, see TEDE)
RG	Regulatory Guide
RPV	reactor pressure vessel
RRV	reactor recirculation valve
RVV	reactor vent valve
scfh	standard cubic feet per hour (unit of flow)
scfm	standard cubic feet per minute (unit of flow)
SG	steam generator
SGTF	steam generator tube failure
SMR	small modular reactor
Sv	sievert (unit of radiation dose)
TEDE	total effective dose equivalent
χ/Q	atmospheric dispersion factor in units of seconds per cubic meter

Table 1-2. Definitions

Term	Definition
Core damage	Assumed to occur at the onset of clad ballooning for the purposes of source term release timing.
Design basis	The entire range of conditions for which a facility is designed in accordance with established design criteria and for which damage to the fuel and release of radioactive material are kept within authorized limits.
Design basis accident	A postulated accident that a nuclear facility must be designed and built to withstand without loss to the systems, structures, and components necessary to ensure public health and safety.
Design basis event	Postulated events used in the design to establish the acceptable performance requirements for the structures, systems, and components.
Design basis source term	Postulated significant core damage event with radionuclides released into an intact containment to enable deterministic evaluation of the response of a facility's engineered safety features.
DE I-131	Dose equivalent I-131 is the concentration of I-131 (μCi/gm) that alone would produce the same dose when inhaled as the combined activities of iodine isotopes I-131, I-132, I-133, I-134, and I-135 present.
DE Xe-133	Dose equivalent XE-133 is the concentration of Xe-133 (μCi/gm) that alone would produce the same acute dose to the whole body as the combined activities of noble gas nuclides Kr-85m, Kr-85, Kr-87, Kr-88, Xe-133, and Xe-135 present.
Loss-of-coolant accident	Those postulated accidents that result in a loss of reactor coolant at a rate in excess of the capability of the reactor makeup system from breaks in the reactor coolant pressure boundary, up to and including a break equivalent in size to the double-ended rupture of the largest pipe of the reactor coolant system.
Maximum hypothetical accident	NuScale follows the approach of the 2012 NEI position paper on SMR source terms (Reference 7.2.20) by referring to the scenario described in footnote 3 of 10 CFR 52.47 (a)(2)(iv) as the maximum hypothetical accident (MHA).
Single release phase	A single release phase of fission products, as opposed to distinct gap release and early in-vessel release phases.
Source Term Design Basis Accident	A postulated accident scenario, meant as a surrogate to the large break LOCA typically evaluated by LWRs to meet the regulatory intent of addressing the MHA, which results in a substantial meltdown of the core with subsequent release of appreciable quantities of fission products into an intact containment.

2.0 Background

In 2012, the Nuclear Energy Institute (NEI) issued a position paper on Small Modular Reactor Source Terms (Reference 7.2.20). NuScale Power, LLC (hereafter, “NuScale”) participated in the development of this position paper. In this document, source terms were divided into two principal categories titled “Category 1” and “Category 2.”

The Category 1 source terms include standard deterministic accidents that are similar to those of large LWRs such as: main steam line break (MSLB), rod ejection accident (REA), fuel handling accident (FHA), steam generator tube failure (SGTF) and small primary coolant line break outside containment. NuScale’s Category 1 source term methodology is consistent with Regulatory Guide (RG) 1.183 methodology. NuScale follows the guidance of RG 1.183 for its approach to Category 1 source terms, except where significant differences exist between the NuScale design and large LWRs. These differences and NuScale’s design specific approach are presented in this report.

The Category 2 source term consists of the maximum hypothetical accident (MHA) scenario in which significant core damage occurs. 10 CFR 52.47 (a)(2)(iv) (Reference 7.2.1) requires nuclear power reactor design certification applicants to evaluate the consequences of a fission product release into the containment assuming the facility is being operated at the maximum licensed power level, and describe what design features are intended to mitigate the radiological consequences of an accident. Footnote 3 of 10 CFR 52.47 (a)(2)(iv) states: “The fission product release assumed for this evaluation should be based upon a major accident, hypothesized for purposes of site analysis or postulated from considerations of possible accidental events. These accidents have generally been assumed to result in substantial meltdown of the core with subsequent release into the containment of appreciable quantities of fission products.” NuScale follows the approach of the 2012 NEI position paper on SMR source terms (Reference 7.2.20) by referring to the scenario described in footnote 3 of 10 CFR 52.47 (a)(2)(iv) as the MHA.

The MHA has historically been linked to a large break LOCA in large LWRs. As described in the NEI position paper (Reference 7.2.20), as SMRs have no large diameter primary coolant system piping, a large break LOCA cannot physically be postulated as the basis for the standard review plan (SRP[1]) Section 15.6.5 analysis of site dose in comparison to 10 CFR 52.47 limits. Therefore, a large break LOCA is not appropriate as the basis for calculating source terms for NuScale. NuScale has developed a methodology to develop surrogate severe accident scenarios, denoted as source term design basis accidents (STDBA), to address the MHA as expressed in SRP 15.6.5 and 10 CFR 52.47 for the off-site and control room doses.

[1] Within the context of this report, reference to the SRP is also meant to account for NuScale design specific review standards (DSRS).

Although the NuScale design may preclude any credible accident scenario that results in substantial core meltdown and fission product release, NuScale recognizes that the analysis of an appropriately determined MHA is necessary to demonstrate that engineered safety features (ESF) provide an acceptable level of protection to the public and control room operators. As stated in RG 1.183, “the design basis accidents were not intended to be actual event sequences, but rather, were intended to be surrogates to enable deterministic evaluation of the response of a facility’s engineered safety features.” NuScale’s methodology utilizes the term DBST for the MHA scenario that meets the intent of 10 CFR 52.47 (a)(2)(iv). As discussed in Section 4.2 of this report, the DBST is composed of a blend of key parameters, such as fuel release fractions and timing, taken from a spectrum of STDBAs.

In addition to the DBST (expected to be presented in Section 15.6.5 of the NuScale DCD), the methodology evaluates design basis accidents (expected to be presented in Section 15.0.3 of the NuScale DCD) for radiological consequences. Table 2-1 is a summary of applicable design basis events for the NuScale design. The table includes the postulated accident sequences, wherein the NuScale design specific review standard (DSRS) or SRP the accident is discussed, what section of RG 1.183 the accident input parameters and assumptions are discussed, and the primary source of radiation for the accident.

Table 2-1. Summary of applicable design basis events to the NuScale design

<table>
<tr><th>Event</th><th>DSRS or SRP</th><th>RG 1.183 Appendix*</th><th>NEI Position Paper Category</th><th>Primary Source of Radiation</th></tr>
<tr><td>DBST (core damage)</td><td>15.0.3</td><td>A</td><td>2</td><td rowspan="3">damaged fuel</td></tr>
<tr><td>Fuel handling accident</td><td>15.7.4</td><td>B</td><td rowspan="5">1</td></tr>
<tr><td>Rod ejection accident</td><td>15.4.8</td><td>H</td></tr>
<tr><td>Main steam line break</td><td>15.1.5</td><td>E</td><td rowspan="3">coolant activity (with iodine spiking)</td></tr>
<tr><td>Steam generator tube failure</td><td>15.6.3</td><td>F</td></tr>
<tr><td>Primary coolant line break</td><td>15.6.2</td><td>n/a</td></tr>
</table>

*Note: Appendices C, D, and G were not included because they are not applicable to the NuScale design.

In addition to the requirements of 10 CFR 52.47 and the guidance of RG 1.183, further guidance is provided in NUREG-1465, RG 1.145, RG 1.194, RG 1.195, and the NuScale DSRS (References 7.2.3-7.2.9).

The NuScale DSRS Section 15.0.3 specifically summarizes the general and specific acceptance criteria for evaluating radiological considerations. These criteria include consideration of atmospheric dispersion and the radiological consequences at the exclusion area boundary (EAB), low population zone (LPZ), control room (CR), and technical support center. Additionally, other accident radiological considerations include

post-accident monitoring and access shielding, among others. NuScale plans to address these considerations separately in the NuScale design certification application.

2.1 Regulatory Requirements

The following regulatory requirements and guidance documents are relevant to the design basis accident radiological evaluations described in this report:

- 10 CFR 50, Appendix A, GDC 19, *Control Room*
- 10 CFR 52.47, *Contents of Applications; Technical Information*
- NUREG-1465, "Accident Source Terms for Light Water Nuclear Power Plants," Revison 0, February 1995
- RG 1.183, "Alternate Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," Revision 0, July 2000
- NuScale Draft DSRS, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for NuScale SMR Design," June 2015
- NuScale Draft DSRS, Section 15.1.5, "Steam System Piping Failures Inside and Outside of Containment," June 2015
- NUREG-0800, Section 15.4.8, "Spectrum of Rod Ejection Accidents (PWR)," Revision 3, March 2007
- NUREG-0800, Section 15.6.2, "Radiological Consequences of the Failure of Small Lines Carrying Primary Coolant Outside Containment," Revision 2, June 1981
- NUREG-0800, Section 15.6.3, "Radiological Consequences of Steam Generator Tube Failure," Revision 2, July 1981
- NUREG-0800, Section 15.7.4, "Radiological Consequences of Fuel Handling Accidents," Revision 1, July 1981
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, February 1983
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," Revision 0, June 2003
- SECY-94-302, "Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs," December 1994

3.0 Methodology Overview

This topical report presents the methodology utilized to perform the radiological calculations associated with Category 1 events and the Category 2 MHA (results are expected to be presented in Section 15.0.3 of the NuScale DCD), with a focus on the following specific methodology, further described in Section 4.0 of this report:

- atmospheric dispersion
- design basis source term
- containment aerosol generation and removal
- post-accident pH_T

Section 3.0 provides a general overview of the methodology and details of industry standard techniques, including software utilized. A flowchart of the radiological consequence calculation process is provided in Figure 3-1, each component of which is discussed in detail in this section.

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Figure 3-1. Flowchart of accident radiological calculation process

3.1 Software

3.1.1 SCALE 6.1/TRITON/ORIGEN-S

SCALE 6.1 modular code package, developed by Oak Ridge National Laboratory, is used for development of reactor core and primary coolant fission product source terms. Specifically, the TRITON and ORIGEN-ARP analysis sequences of the SCALE 6.1 modular code package, and ORIGEN-S, run as a standalone module, are used to generate radiation source terms for the NuScale fuel assemblies and primary coolant (Reference 7.2.25). The aforementioned software has been used in the evaluation of operating large LWRs. The operating environment, nuclear fuel and structural materials in the NuScale design are expected to be similar to, or bounded by, that in large pressurized water reactors (PWR).

3.1.1.1 TRITON

As described in the SCALE manual (Reference 7.2.25), the TRITON sequence of the SCALE code package is a multipurpose control module for nuclide transport and depletion, including sensitivity and uncertainty analysis. TRITON can be used to generate problem- and exposure-dependent cross sections as well as perform multi-group transport calculations in one-dimensional, two-dimensional, or three-dimensional geometries. The ability of TRITON to model complex fuel assembly designs improves transport modeling accuracy in problems that have a spatial dependence on the neutron flux. In this case, TRITON is used to generate burnup-dependent cross sections for NuScale fuel assemblies for subsequent use in the ORIGEN-ARP depletion module.

3.1.1.2 ORIGEN (ORIGEN-ARP and ORIGEN-S)

Reference 7.2.25 describes ORIGEN-ARP as a SCALE depletion analysis sequence used to perform point-depletion and decay calculations with the ORIGEN-S module using problem- and burnup-dependent cross sections. ORIGEN-S nuclear data libraries containing these cross sections are prepared by the ARP module using interpolation in enrichment and burnup between pre-generated nuclear data libraries containing cross section data that span the desired range of fuel properties and operating conditions. The ORIGEN-ARP sequence produces calculations with accuracy comparable to that of the TRITON sequence with a savings in problem setup and computational time as compared to repeated use of TRITON. Many variations in fuel assembly irradiation history can be modeled. For depletion calculations involving NuScale fuel assemblies, the ORIGEN-S nuclear data libraries are generated by the TRITON sequence, as described in the previous Section 3.1.1.1.

3.1.2 ARCON96

The calculation of both onsite and offsite atmospheric dispersion factors for design basis accidents is performed with ARCON96 (Reference 7.2.24). The program implements the guidance provided in RG 1.194 (Reference 7.2.8). The code implements a building wake dispersion algorithm; an assessment of ground level, building vent, elevated and diffuse source release modes; use of hour-by-hour meteorological observations; sector

averaging and directional dependence of dispersion conditions. The code also implements a Gaussian diffusion model for the 0 to 8 hour period.

NuScale uses ARCON96 for various time periods at the EAB and the outer boundary of the LPZ as well as the control room and technical support center. Justification for utilizing ARCON96 for offsite locations, as opposed to PAVAN, is provided in Section 4.1.

3.1.3 RADTRAD

RADTRAD is used to estimate radionuclide transport and removal of radionuclides and dose at selected receptors for the various DBAs (Reference 7.2.31). Given the radionuclide inventory, release fractions and timing, RADTRAD estimates doses at offsite locations, i.e., the EAB and LPZ, and inside the control room and technical support center. As material is transported through the containment, the user can account for natural deposition that may reduce the quantity of radioactive material. Material can flow between buildings, from buildings to the environment, or into the control rooms through filters, piping or other connectors. An accounting of the amount of radioactive material retained due to these pathways is maintained. Decay and in-growth of daughters can be calculated over time as material is transported.

3.1.4 MELCOR

MELCOR is used to model the progression of severe accidents through modeling the major systems of the plant and their generally coupled interactions (Reference 7.2.13). Specific use relevant to the application of DBST includes the following:

- thermal-hydraulic response of the primary coolant system and containment vessel
- core uncovering, fuel heatup, cladding oxidation, fuel degradation and core material melting and relocation
- aerosol generation
- in-vessel and ex-vessel hydrogen production and transport
- fission product release (aerosol and vapor) and transport
- and impact of engineered safety features on thermal-hydraulic and radionuclide behavior

3.1.5 NRELAP5

NRELAP5 is NuScale’s proprietary system thermal-hydraulic computer code used in engineering design and analysis. It has been developed for best-estimate transient simulation of LWR coolant systems during postulated accidents. The code models the coupled behavior of the reactor coolant system and the core for LOCAs and operational transients. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers, and secondary feedwater systems. NRELAP5 is developed at NuScale, with RELAP5-3D© v.4.1.3 as

the initial baseline. RELAP5-3D© v.4.1.3 was procured from the Idaho National Laboratory through a commercial grade dedication process. Upon dedication, the RELAP5-3D© v.4.1.3 code was renamed to NRELAP5 and further developed by NuScale.

3.1.6 STARNAUA

Aerosol transport and removal calculations are provided by the program STARNAUA. STARNAUA is an aerosol transport and removal software program that was developed by Polestar Applied Technology, Inc., a company later purchased by WorleyParsons. STARNAUA is an enhanced version of NAUAHYGROS and was developed by Polestar for performing aerosol removal calculations in support of work to develop and apply a realistic source term for advanced and operating LWRs.

It models natural removal of containment aerosols by gravitational settling and diffusiophoresis, and considers the effect of hygroscopicity (growth of hygroscopic aerosols due to steam condensation on the aerosol particles) on aerosol removal. In developing STARNAUA, Polestar enhanced NAUAHYGROS by adding a model for thermophoresis, a model for spray removal, and the capability to directly input steam condensation rate or condensation heat transfer rate, and total heat transfer rate such as would be provided from an external containment thermal hydraulics code calculation.

This software was developed for the purpose of performing aerosol removal calculations to apply in realistic source terms for advanced and operating LWRs. This realistic source term methodology is consistent with existing industry practice used for large passive LWR design certification.

3.1.7 pH_T

The Fortran program developed by NuScale to calculate post-accident aqueous molar concentration of hydrogen ions (pH_T) is called “pH_T”. This program calculates pH_T utilizing the methodology described in Section 4.5. This program takes inputs for initial boron and lithium concentrations, the total core inventory of iodine and cesium, the integrated photon dose to the containment and total dose to the coolant, the initial mass of coolant, the mass of coolant, and the temperature of the coolant. The program then calculates the coolant pH_T as a function of time.

3.1.8 MCNP

MCNP is utilized for evaluating potential shine radiological exposures, or doses, to operators within the control room following a radiological release event. Direct shine, sky-shine, and shine from all possible filters are evaluated. MCNP is a general-purpose tool used for neutron, photon, electron, or coupled neutron, photon, and electron transport (Reference 7.2.27). MCNP treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. The code is well-suited to performing fixed source calculations of the type documented herein.

MCNP uses continuous energy cross-section data. For photons, the code accounts for incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, and absorption in electron-positron pair production. Electron and positron transport processes account for angular deflection through multiple Coulomb scattering, collisional energy loss with optional straggling, and the production of secondary particles including x-rays, knock-on and Auger electrons, bremsstrahlung, and annihilation gamma rays from positron annihilation at rest.

3.2 Overview of Category 1 Events

3.2.1 Rod Ejection Accident

NuScale plans to provide in the design certification application the final results, with radiological consequence analysis if applicable, of the NuScale REA. NuScale utilizes the REA methodology guidance enumerated in Appendix H of RG 1.183. Appendix H of RG 1.183 states that no radiological consequences analysis is required if no fuel damage is indicated in the analysis and the accident is bounded by other events. {{

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Containment release path:

1. A control rod ejection occurs, resulting in a rapid positive reactivity insertion.
2. A portion of the fuel rods are damaged by either cladding breach or melt failure modes.
3. All of the activity released from the fuel is instantaneously and homogenously mixed in the containment atmosphere.
4. The containment leaks as described in Section 3.3.7.

Primary system release path:

1. A control rod ejection occurs, resulting in a rapid positive reactivity insertion.
2. A portion of the fuel rods are damaged by either cladding breach or melt failure modes.
3. All of the activity released from the fuel is instantaneously and homogenously mixed in the primary system.
4. Primary coolant leaks into both steam generators at the maximum leak rate allowed by design basis limits. The leakage continues until the reactor is shut down and depressurized and the primary and secondary systems are at an equal pressure.
5. Activity is released to the environment through the condenser until isolation is achieved.

6. Leakage through the secondary isolation valves (main steam and feedwater) occurs in the reactor building until the reactor is shut down and depressurized. No credit is taken for any source term reduction within the reactor building.

The following is a summary of the assumptions used from Appendix H of RG 1.183:

- containment iodine chemical form of 95% cesium iodide, 4.85% elemental iodine, and 0.15% organic iodide
- primary system iodine chemical form of 97% elemental iodine and 3% organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system
- density for leak rate conversion: 62.4 pound mass (lbm)/ft^3

3.2.2 Fuel Handling Accident

The methodology for determining FHA radiological consequences is based on the guidance provided in Appendix B of RG 1.183 and Section 15.7.4 of the SRP. The explicit guidance enumerated in Appendix B of RG 1.183 is followed with one exception, which is that the iodine decontamination factor will be calculated with a generalized methodology instead of utilizing the prescribed RG 1.183 values for a depth of water above the damaged fuel of 23 feet or greater. The methodology assumes failure of {{ }}$^{2(a),(c)}$ occurs.

As presented in Section 3.3.8 of this report, the NuScale reactor pool has a minimum depth above the damaged fuel greater than the minimum 23 foot depth specified as the basis for the iodine decontamination factor in Reference 7.2.11. Therefore, a generalized methodology for calculating increased decontamination factor was used, and is based on the methodology and assumptions of Reference 7.2.12. This methodology is presented in more detail in Section 3.3.8.

The following is a summary of the assumptions used from Appendix B of RG 1.183:

- radionuclides considered include xenon, krypton, halogen, cesium, and rubidium
- iodine chemical form of 57 percent elemental iodine and 43 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the fuel
- release to the environment over a two hour period

The standard activity release period of two hours is used for the dose assessment. This period is the standard assumption provided in Section 4.1 of Appendix B of RG 1.183, which states that “For fuel handling accidents postulated to occur within the fuel building, the following assumptions are acceptable to the NRC staff: The radioactive material that escapes from the fuel pool to the fuel building is assumed to be released to the environment over a 2-hour time period.”

3.2.3 Main Steam Line Break outside Containment

Radiological consequences of the main steam line break outside containment accident are calculated based on the guidance provided in Appendix E of RG 1.183. The NuScale methodology for calculating the radiological consequences of this event follow the explicit guidance enumerated in Appendix E of RG 1.183.

This radiological consequence analysis considers the main steam line break event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows:

1. A main steam line break occurs in one of the two main steam lines.
2. For each of the iodine spiking scenarios, the iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by primary coolant system design basis limits.
3. Primary coolant leaks into the secondary side of the intact steam generators at the maximum leak rate allowed by design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.
4. A time-dependent release is modeled that effectively releases the activity directly to the environment through the break.
5. The non-faulted steam line continues to release a small quantity of radiation through valve leakage.

The following is a summary of the assumptions used from Appendix E of RG 1.183:

- coincident iodine spiking factor: 500
- duration of coincident iodine spike: 8 hr
- density for leak rate conversion: 62.4 lbm/ft^3
- iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system

3.2.4 Steam Generator Tube Failure

Radiological consequences of the steam generator tube failure accident are calculated based on the guidance provided in Appendix F of RG 1.183 and Section 15.6.3 of the SRP. The NuScale methodology for calculating the radiological consequences of this event follows the explicit guidance enumerated in Appendix F of RG 1.183.

This radiological consequence analysis considers the steam generator tube failure event with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A description of the scenario evaluated is summarized as follows:

1. A steam generator tube failure occurs in one of the two steam generators.
2. For each of the iodine spiking scenarios, the iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits.
3. Primary coolant flows into the secondary coolant through the failed steam generator tube at a rate and duration defined by the transient analysis.
4. Primary coolant leaks into the secondary side of the intact steam generators at the maximum leak rate allowed by primary coolant system design basis limits. The leakage continues until the primary system pressure is less than the secondary system pressure.
5. A time-dependent release is modeled that effectively releases the activity directly to the environment through the break.
6. Once secondary system isolation occurs, both steam lines continue to release small quantities of radiation through valve leakage into the reactor building which is assumed to flow directly into the environment without any source term reduction.

The following is a summary of the explicit assumptions used from Appendix F of RG 1.183:

- coincident iodine spiking factor: 335
- duration of coincident iodine spike: 8 hr
- density for leak rate conversion: 62.4 lbm/ft^3
- iodine chemical form: 97 percent elemental iodine and 3 percent organic iodide
- no reduction or mitigation of noble gas radionuclides released from the primary system

3.2.5 Failure of Small Lines Carrying Primary Coolant outside Containment

Failure of small lines carrying primary coolant outside containment is not an event specifically addressed in RG 1.183 and Section 15.6.2 of the SRP only provides general guidance for this event. Therefore, the methodology, including the iodine spiking assumptions, developed by NuScale for this event is similar to the main steam line break and steam generator tube failure. An event-specific transient analysis is used to define the time-dependent release of activity.

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3.3 General Methodology and Assumptions

3.3.1 Core Radionuclide Inventory

In order to establish the amount of radionuclides that could be released, the reactor core radionuclide inventory must be established.

The isotopic inventories of fuel assemblies are calculated using the SCALE 6.1 package described in Section 3.1.1. This methodology includes an assumption for maximum activity for each isotope and is used in the radiological consequence analysis. Isotopic concentrations are based on the detailed geometry of a fuel assembly, rated power plus uncertainty, maximum possible assembly average exposure, and a range of U^{235} enrichments. Table 3-1 includes a summary of example parameters that could be used for determining radionuclide inventory. The isotopic inventory is calculated at a number of time steps in the fuel cycle, the number of which is calculated based on the recommendations of the modeling guidelines of Reference 7.2.25. For each isotope, the maximum curie content of any time step is used as the activity for that isotope at the beginning of any event.

Table 3-1. Example NuScale parameters for core radionuclide inventory

Description	Example Value
Core Power Uncertainty	2 percent
Assembly Average Exposure	{{ }}2(a),(c)-ECI
Evaluated U^{235} enrichments range	1.5 percent - 5 percent
Number of time steps in fuel cycle	40

For each radiological consequence calculation in which the fuel assembly is the source term, the activity of a single fuel assembly ($A_i^{Assembly}$) for each isotope may then be utilized. The total released activity ($A_i^{Released}$) is determined by the release fractions defined for the event and isotope, expressed as

$$A_i^{\text{Released}} = A_i^{Assembly} \times \begin{pmatrix}\text{Radial}\\ \text{Peaking}\end{pmatrix} \times \begin{pmatrix}\text{Number of}\\ \text{Assemblies}\end{pmatrix} \times \begin{pmatrix}\text{Release}\\ \text{Fraction}\end{pmatrix}_i \Big/ \begin{pmatrix}\text{Decon.}\\ \text{Factor}\end{pmatrix}_i \qquad \text{Eq 3-1}$$

Two options ("deterministic" and "as-loaded") are provided for modeling the total irradiation history of the evaluated fuel assemblies in a reactor core; either of which are acceptable. Each specific application referencing this methodology must specify which option is utilized.

A "deterministic" option is a treatment of exposure such that it is assumed that all fuel assemblies in the core are irradiated to the maximum allowed assembly exposure {{ }}2(a),(c)-ECI as an example). Thus, a full core inventory is a simplified "single batch" core design used for analysis purposes in which it is assumed that the fuel is irradiated at constant full power plus uncertainty until the maximum exposure is reached.

This is an alternative to an "as-loaded" option, which is a best-estimate approach with respect to uniquely loaded fuel batches. With this option, isotopic inventories are given at the end of irradiation for several fuel assembly types, for each cycle (a multi-batch core design). In this approach, a whole-core inventory is calculated through the weighted sum of the values for each fuel assembly type. As noted above, each specific application that references this methodology must specify in its analyses which option was utilized.

3.3.2 Primary Coolant Radionuclide Inventory

For the radiological consequence analysis, the radioactive concentrations in the primary coolant system are set at the maximum dose equivalent values permitted by design basis limits. {{

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3.3.2.1 Secondary Coolant Activity

Large PWR designs contain a large volume of secondary system water on the “shell” side of the steam generator heat exchanger. Through primary-to-secondary leakage limits and monitoring by sampling, this water volume contains levels of iodine that are limited operationally. The NuScale design is the opposite, in that the “shell” side of the heat exchanger is the primary coolant and the “tube” side is the secondary coolant. {{

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3.3.3 General Dose Analysis Inputs

The dose analysis program RADTRAD uses a combination of tables and numerical models of source term transport phenomena to determine the time-dependent dose at user-specified locations for a given accident scenario. The model also provides the decay chain and dose conversion factor tables needed for the dose calculation. The user

provides the atmospheric relative concentrations *X/Q* for offsite locations and the control room. In addition, the breathing rates and the control room occupancy factors are provided by the user.

3.3.3.1 Source Term Release Fraction and Timing for Dose Analysis

Isotopic activities derived for input to radiological consequence analysis are described in Section 3.3.1. Section 3.3.1 also describes how release fraction and timing effects for RADTRAD calculations are addressed by factoring the core radionuclide inventory with release fractions. A detailed description of fission product release fraction and timing information generated from MELCOR is provided in Section 4.2.3.

3.3.3.2 Atmospheric Dispersion Factors X/Q, Breathing Rates, and Occupancy Factors

Atmospheric dispersion factor *X/Q* inputs to RADTRAD are derived as described in Section 5.1. Control room and offsite breathing rate inputs to RADTRAD, consistent with RG 1.183 (Reference 7.2.2), are summarized in Table 3-2.

Table 3-2. Offsite and control room breathing rates (m^3/sec)

Time (Hr)	CR	EAB	LPZ
0 - 8	3.50E-04	3.50E-04	3.50E-04
8 - 24		1.80E-04	1.80E-04
24 - 720		2.30E-04	2.30E-04

Control room occupancy factor inputs to RADTRAD, consistent with RG 1.183 (Reference 7.2.2), are summarized in Table 3-3.

Table 3-3. Control room occupancy factors

Time (Hr)	Occupancy (percent)
0 - 8	100
8 - 24	60
24 - 720	40

3.3.3.3 Flow Rates

Values assumed for the tabular flow rate and timing inputs to RADTRAD are based on habitability system capacities. The final values are expected to be evaluated in the design certification application.

3.3.3.4 Dose Conversion Factors

Consistent with RG 1.183 (Reference 7.2.2), dose conversion factors from Environmental Protection Agency (EPA) Federal Guidance Report No. 11 and EPA

Federal Guidance Report No. 12 (References 7.2.38 and 7.2.39, respectively) are used for dose analysis.

3.3.4 General Dose Analysis Assumptions

3.3.4.1 Control Room Ventilation Design

NuScale plans to provide the final control room ventilation design with the NuScale design certification application. A representative design for the control room ventilation was used to confirm the methodology assumptions. The key design features assumed for this methodology are summarized as follows:

- The nonsafety related normal control room ventilation is isolated by a safety-related control system once a sufficiently high source of radioactivity is measured at the duct intake
- An emergency source of pressurized air provides clean air for 72 hours
- After 72 hours of emergency operation, the normal control room ventilation system is used again
- The control room is habitable during a loss of offsite power as the emergency mode will be automatically activated if this occurs
- Control ventilation is designed to minimize in-leakage

For the example calculations provided in Section 5.0 of this report, the following modeling assumptions are defined in Table 3-4.

Table 3-4. Example control room characteristics

Description	Units	Value
Control Room Isolation Time	min	{{
Control Room Envelope Volume	ft^3	
Control Room Emergency Flow Rate	cfm	
Control Room Normal Flow Rate	cfm	
Control Room Emergency Duration	hr	
Control Room Unfiltered Ingress/Egress	cfm	
Control Room Unfiltered In-leakage	cfm	}}2(a),(c)

3.3.4.2 Control Room Dose Mitigation Equipment

No credit is taken for the use of personal protective equipment such as protective beta radiation resistant clothing, eye protection, or self-contained breathing apparatus. Similarly, no credit is taken for prophylactic drugs such as potassium iodide pills.

3.3.4.3 Reactor Building Decontamination

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3.3.4.4 Pre-Accident Coolant Radiation Levels

For events in which the source of radiation is from damaged fuel, it is assumed that the primary and secondary coolant radiation levels are zero prior to the accident level with the exception of pre-incident iodine spiking scenarios required for evaluation. This assumption is in accordance with RG 1.183 (Reference 7.2.2) that prescribes the source term assumptions and isotopes to be used for each event. In particular, the fuel releases are assumed to occur and mix instantaneously within the reactor coolant system. For events in which the source of radiation is from primary coolant, the primary coolant radiation levels prescribed by RG 1.183 are utilized.

3.3.4.5 Control Room Exhaust

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3.3.4.6 Radiation Shine Radiological Consequences

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3.3.4.7 Reactor Building Pool Boiling Radiological Consequences

An extended loss-of-offsite power event is expected to result in the decay heat from the reactors and the spent fuel to heat up the spent fuel pool and eventually cause the reactor pool to boil. The dose contribution of the pool boiling, such as would be postulated to occur in the NuScale reactor pool in the event of an extended loss of power to the pool heat removal system, is accounted for in the following manner.

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3.3.4.8 Parent and Daughter Isotopes

Consistent with RG 1.183 (Reference 7.2.2), the RADTRAD decay and daughtering modeling option is used to include progeny from the decay of parent radionuclides that are significant with regard to radiological consequences and the released radioactivity. The calculated total effective dose equivalent (TEDE) dose is thus the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure from all tracked isotopes.

3.3.4.9 Two-Hour Sliding Window

RADTRAD determines the maximum two-hour TEDE by calculating the postulated dose for a series of small time increments and performing a "sliding" sum over the increments of successive two-hour periods. The time increments appropriately reflect the progression of the accident to capture the peak dose interval between the start of the event and the end of radioactivity release.

3.3.4.10 Effluent Plume Depletion

Consistent with RG 1.183 (Reference 7.2.2), the RADTRAD model does not include corrections for depletion of the effluent plume by deposition on the ground.

3.3.4.11 Direct Release Path

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3.3.5 Offsite Dose Calculation

As defined in Reference 7.2.31, the dose to a hypothetical individual is calculated with RADTRAD using user specified *X/Qs* and the amount of each nuclide released during the exposure period. The air immersion dose from each nuclide, n, in an environmental compartment is calculated as:

$$D_{c,n}^{env} = A_n(X/Q)DCF_{c,n} \qquad \text{Eq 3-12}$$

where

$D_{c,n}^{env}$ = air immersion (cloudshine) dose due to nuclide n in the environment compartment (Sievert (Sv))

$DCF_{c,n}$ = Federal Guidance Report No. 11 and Federal Guidance Report No. 12 air immersion (cloudshine) dose conversion factor for nuclide n as discussed in Section 3.3.3.4 (Sv·m^3/Becquerel(Bq)·s)

$\mathrm{X/Q}$ = user-provided atmospheric relative concentration (s/m^3)

$\mathrm{A_n}$ = released activity of nuclide n (Bq)

The activity is related to the number of atoms of nuclide n as:

$$A_n = N_n\lambda_n \qquad \text{Eq 3-13}$$

Where λ_n is the radiological decay constant for the nuclide.

The inhalation dose from each nuclide, n, is calculated as:

$$D_{i,n}^{env} = A_n(X/Q)BRDCF_{i,n} \qquad \text{Eq 3-14}$$

where

$D_{i,n}^{env}$ = inhalation dose commitment due to nuclide n in the environment compartment (Sv)

BR = user-provided breathing rate (m^3/s)

$\mathrm{DCF_{i,n}}$ = user-provided inhalation dose conversion factor for nuclide n as discussed in Section 3.3.3.4 (Sv/Bq)

3.3.6 Control Room Dose Calculation

Per Reference 7.2.31, control room dose is calculated with RADTRAD based on the time-integrated concentration in the control room compartment using the user input atmospheric dispersion factors and breathing rates. The air immersion dose in the control room is calculated as:

$$D_{c,n}^{CR} = \int C_n(t)dt\left(DCF_{c,n}/G_F\right) \times OF \qquad \text{Eq 3-15}$$

where $C_n(t)$ is the instantaneous concentration of radionuclide n in the compartment and OF occupancy factor. The Murphy–Campe (Reference 7.2.58) geometric factor G_F relates the dose from an infinite cloud to the dose from a cloud of volume V as:

$$G_F = \frac{1173}{V^{0.338}} \qquad \text{Eq 3-16}$$

The inhalation dose in the control room is:

$$D_{i,n}^{CR} = \int C(t)dt(BR \cdot OF \cdot DCF) \qquad \text{Eq 3-17}$$

3.3.7 Containment Leakage

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3.3.8 Fuel Handling Accident Decontamination

The methodology for determining the radiological consequences of a FHA assumes that the NuScale reactor pool (or spent fuel pool depending on the location of the FHA) has a minimum water depth above the damaged fuel greater than the 23-foot depth specified in RG 1.183. In accordance with RG 1.183, the guidance of Reference 7.2.12 is utilized to establish a NuScale specific reactor pool decontamination factor for the FHA.

Page 26 of Reference 7.2.12 defines the pool inorganic decontamination factor to be proportional to an exponential function with the pool depth in the exponent as given by

$$DF_{inorg} = e^{\frac{6k_{eff}H}{d_b v_b}} \qquad \text{Eq 3-18}$$

where,

d_b = Diameter of bubble
DF_{eff} = Effective decontamination factor for iodine
DF_{inorg} = Decontamination factor for inorganic iodine
F_{inorg} = Fraction of inorganic iodine
F_{org} = Fraction of organic iodine
H = Height of bubble rise (i.e., bubble rise height)

k_{eff} = Effective flow characteristics of bubble

v_b = Rise velocity of a bubble from pressurized source

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Table 3-5. Comparison of original RG 1.183 values and example effective decontamination factor scaled to varying water depths

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3.3.9 Iodine Spiking

The NRC's results of the initial screening of Generic Issue (GI) 197 (Reference 7.2.34) describes the phenomenon of iodine spiking observed in operating reactors. After a core power or primary system pressure transient, the iodine concentration in the reactor coolant may increase to a value many times its equilibrium concentration level, followed by a gradual decay back down to a lower level. Iodine spiking occurs when a change in reactor power, temperature, and/or pressure results in the transport of dissolved iodine compounds out of failed fuel rods and into the primary coolant. After reaching peak concentrations, the iodine is then gradually removed by the reactor coolant cleanup systems, radioactive decay, and release to the environment.

All known iodine spiking models are built on an assumed physical causative scenario of a fuel rod with a defect. During power operation, iodine collects on the surfaces of the fuel pellets and internal cladding surface; likely as cesium iodide or another water-soluble salt. However, during operation, the internal free volume of the defective fuel rod is steam-blanketed, and relatively little iodine is transported out to the reactor coolant. If the reactor is shut down, or if power is reduced in a power transient, liquid water will enter the fuel pellet-to-cladding gap volume, dissolving any soluble iodine compounds, which then can readily diffuse out of the cladding defect. Similarly, a pressure transient could force liquid water in or out of the defective fuel rod, thereby transporting iodine into the bulk primary coolant.

It should be noted that, if there were no cladding defects in the core, then according to this model the specific activity of iodine on the cladding surface would drop to zero, under both equilibrium and non-equilibrium conditions. The presence of traces of uranium on the outside of the cladding left over from manufacture of the fuel, complicates the model. Iodine produced from fission of a trace uranium atom would not be expected to contribute to spiking, since it is already outside of the cladding, but would contribute to the equilibrium specific activity in the coolant.

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3.3.10 Steam Generator Decontamination

The helical coil steam generators of the NuScale design are different than that of a large PWR because the primary coolant is on the outside of the steam generator tubes. As a result, there is not a bulk water volume in which decontamination could easily occur. {{

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3.3.11 Removal in Piping and Main Condenser

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4.0 NuScale Unique Methodology

4.1 Atmospheric Dispersion

NuScale plans to postulate in the NuScale design certification application an exclusion area boundary (EAB) and low population zone (LPZ) at the site boundary, which is estimated to be in the range of 80 to 400 meters. This range of distances is shorter than the boundaries associated with standard nuclear power plants, which may range from 800-6000 meters. This postulated LPZ and EAB for the NuScale design certification application is expected to be in the near-vicinity of buildings. For these conditions, in which the LPZ is near buildings, the industry-standard methodology for the calculation of offsite relative dose concentrations is less robust than the methodology for calculation of onsite relative dose concentrations.

The methodology presented in this report uses ARCON96 for offsite and control room radiological consequence analyses. The PAVAN methodology, based upon the guidance presented in RG 1.145 (Reference 7.2.7), over-predicts relative dose concentrations in the vicinity of buildings for the limiting case. This conclusion is consistent with Reference 7.2.23, which states:

> In the mid-1980s, the staff of the U.S. Nuclear Regulatory Commission (NRC) felt that its guidance to licensees related to calculating atmospheric concentrations of radionuclides and toxic chemicals in the vicinity of buildings was overly conservative.

ARCON96 methodology, based on RG 1.194 (Reference 7.2.8), is more accurate than PAVAN for predicting atmospheric dispersions in the vicinity of buildings for the limiting case, while still producing predictions with conservative margins. As illustrated in Sections 4.1.3 and 4.1.4, this difference in accuracy is most evident at shorter distances. This reasoning was the basis of the development of ARCON96. The model's purposes are directly relevant to the NuScale offsite relative concentration calculation, as described in the following sections.

4.1.1 PAVAN

Detailed information regarding PAVAN methodology can be found in the PAVAN User's Manual (Reference 7.2.22). PAVAN methodology is based upon the guidance presented in RG 1.145 (Reference 7.2.7), which describes four regulatory positions. Position one addresses the calculation of relative concentrations, position two addresses the determination of maximum sector relative concentrations, position three addresses the determination of a five percent overall site relative concentration, and position four concerns the selection of relative concentrations to be used in evaluations. The following is a summary of this guidance.

Position one: The meteorological data needed for relative concentration calculations include a joint frequency distribution (JFD) of hourly wind speed, wind direction, and a measure of atmospheric stability for one year. A consecutive 24-month period of onsite meteorological data is expected to be included in an early site permit (ESP) or combined

license (COL) application that does not reference an ESP per SRP Section 2.3.3 (Reference 7.2.14) and RG 1.23 (Reference 7.2.6). Wind direction is classed into 16 separate 22.5-degree sectors. Two-hour relative concentrations are calculated through selective use of Eq 4-1, Eq 4-2, and Eq 4-3 by assuming meteorological data representing 1-hour averages are applicable to the 2-hour period. Eq 4-1 and Eq 4-2 are used to account for building wake effects, and Eq 4-3 is used to account for plume meander. The maximum relative concentration calculated from Eq 4-1 and Eq 4-2 is compared with the relative concentration calculated from Eq 4-3, and the minimum is selected.

$$\frac{X}{Q}(x,i,j) = \left\{U_{ij}(10)\left[\pi\sigma_{yj}(x)\sigma_{zj}(x) + cA\right]\right\}^{-1} \qquad \text{Eq 4-1}$$

$$\frac{X}{Q}(x,i,j) = \left\{3U_{ij}(10)\pi\sigma_{yj}(x)\sigma_{zj}(x)\right\}^{-1} \qquad \text{Eq 4-2}$$

$$\frac{X}{Q}(x,i,j) = \left\{U_{ij}(10)\pi M_{ij}(x)\sigma_{yj}(x)\sigma_{zj}(x)\right\}^{-1} \qquad \text{Eq 4-3}$$

$\frac{X}{Q}(x,i,j)$	= relative concentration
x	= downwind distance (meters)
i	= wind-speed category
j	= stability category
$\sigma_{yj}(x)$	= lateral dispersion of plume for stability category j at distance x
$\sigma_{zj}(x)$	= vertical dispersion of plume for stability category j at distance x
c	= mixing volume coefficient in building-wake term (set to 0.5)
A	= minimum cross-sectional area of the building
$M_{ij}(x)$	= meander factor for lateral plume spread
$U_{ij}(10)$	= adjusted average wind speed for wind speed and stability

$$M_{ij}(x)\sigma_{yj}(x) = \sigma_{yj}(x) + \left[M_{ij}(x) - 1\right]\sigma_{yj}(800) \qquad \text{Eq 4-4}$$

[Note: The $M_{ij}(x)\ \sigma_{yj}(x)$ term from Eq 4-3 is redefined in Eq 4-4 for downwind distances greater than 800 meters. For downwind distances less than 800 meters, Eq 4-4 is not used.]

$\sigma_{yj}(800)$ = lateral dispersion of plume at 800 meters

Two-hour relative concentrations are calculated for EAB and LPZ distances for each hour of data by assuming meteorological data representing 1-hour averages are applicable to the 2-hour period. An annual average is also calculated for each sector at the LPZ distance and is used in combination with the two-hour relative concentration in order to determine relative concentrations for various intermediate time periods.

Position two: Using relative concentrations calculated for each hour of data, a cumulative probability distribution of relative concentrations is constructed for each of the 16 sectors. A plot of relative concentration versus probability of being exceeded is made for each sector and a smooth curve is drawn to form an upper bound of the computed points. For each of the 16 curves, the relative concentration that is exceeded 0.5 percent of the total number of hours in the data set should be selected. The highest of the 16 sector values is defined as the maximum sector *X/Q*. Maximum sector relative concentrations are calculated for the 0 to 2 hour time period for the EAB. Maximum sector relative concentration for the 0 to 2 hour time period and the intermediate time periods are calculated for the LPZ.

Position three: Using relative concentrations calculated for each hour of data, an overall cumulative probability distribution for all directions combined is constructed. A plot of relative concentration versus probability of being exceeded is made, and an upper bound curve is drawn. The two-hour relative concentration that is exceeded five percent of the time should be selected from this curve. In addition, for the LPZ distance, the maximum of the 16 annual average relative concentrations should be used along with the five percent two-hour relative concentration to determine relative concentrations for the intermediate time periods.

Position four: The relative concentration for EAB or LPZ distances should be the maximum sector *X/Q* (position two) or the 5 percent over site *X/Q* (position three), whichever is higher.

4.1.2 ARCON96

Detailed information regarding ARCON96 methodology and a description of the technical basis for the code is provided in Reference 7.2.24. The following paragraphs provide a brief summary of relevant sections of this technical basis and information from RG 1.194 (Reference 7.2.8).

The meteorological data needed for relative concentration calculations include hourly data of wind speed, wind direction, and a measure of atmospheric stability for one year. A consecutive 24-month period of onsite meteorological data is expected to be included in an ESP or COL application that does not reference an early site permit per SRP

Section 2.3.3 and RG 1.23. Relative concentrations are calculated for each hour through use of Eq 4-5 and Eq 4-6. ARCON96 estimates diffusion in building wakes by replacing the σ_y and σ_z terms in Eq 4-5 with the Σ_y and Σ_z terms in Eq 4-6.

The subscript y indicates horizontal direction and the subscript z indicates the vertical direction.

$\Delta\sigma_1$: (the low wind speed increment) is the factor that accounts for plume meander.

$\Delta\sigma_2$: (the high wind speed increment) is the factor that accounts for building wake effects, and σ is the normal diffusion coefficient.

y is the distance from the center of the plume

$$\frac{\chi}{Q'} = \frac{1}{\pi\sigma_y\sigma_z U} exp\left[-0.5\left(\frac{y}{\sigma_y}\right)^2\right] \qquad \text{Eq 4-5}$$

$$\Sigma_y = \sqrt{\sigma_y^2 + \sigma_{y1}^2 + \sigma_{y2}^2}$$
$$\Sigma_z = \sqrt{\sigma_z^2 + \sigma_{z1}^2 + \sigma_{z2}^2} \qquad \text{Eq 4-6}$$

Intermediate time periods are calculated using different averages of each hourly relative concentration. A cumulative frequency distribution is constructed for each averaging period, and the 95th percentile relative concentration is selected from each, using linear interpolation. These relative concentrations are used to calculate the 95th percentile relative concentration for each standard averaging interval.

4.1.3 Major Differences

The following list summarizes the key differences between PAVAN and ARCON96 program methodology, using the information described in Sections 4.1.1 and 4.1.2 of this report.

- Generally, PAVAN uses a JFD of hourly wind speed, wind direction, and a measurement of stability class, while ARCON96 uses hourly data.
- PAVAN relies upon selective use of three different equations to account for plume meander and building wake effects, while ARCON96 relies upon one equation that accounts for both factors as a function of wind speed.
- PAVAN calculates a 99.5th percentile relative concentration for each sector and a 95th percentile relative concentration for the site limit, while ARCON96 only calculates a 95th percentile relative concentration.

- PAVAN calculates a relative concentration for each of the 16 direction sectors with only one execution of the code, while ARCON96 calculates a relative concentration for one specified direction sector per code execution. The direction sector can be specified in any direction from the intake to the source when executing ARCON96. NuScale utilizes 16 different 22.5 degree direction sectors for ARCON96 to be consistent with PAVAN, which utilizes 16 direction sectors that are each 22.5 degrees.
- PAVAN assumes a default direction window of 22.5 degrees, while ARCON96 allows a custom input direction window. The default direction window input for ARCON96 is 90 degrees; NuScale's methodology is to utilize this default 90 degree setting.

As stated above, ARCON96 calculates relative concentrations in one of 16 possible direction sectors at a time, while PAVAN calculates relative concentrations for all 16 direction sectors. Therefore, in order to use ARCON96 for offsite purposes, 16 executions of the code must be performed (one for each direction sector). The NuScale methodology for the use of ARCON96 for offsite purposes assumes a uniform circle where each of the 16 direction sectors is of equal length. Each execution of ARCON96 produces a 95th percentile relative concentration for one direction sector. The maximum relative concentration found in each set of 16 code executions for a particular distance and time period should be selected as the relative concentration to utilize for dose consequence analysis associated with the particular distance and time period.

ARCON96 does not calculate a maximum sector 99.5th percentile as performed by PAVAN and stated in RG 1.145 (Reference 7.2.7) regulatory position 2. However, the ARCON96 methodology utilizing the 95th percentile value has been approved for control room calculations, which is utilized as precedent for development of the methodology described in this report, due to the similar distances assumed in the methodology.

With respect to the ARCON96 direction window input, the default 90 degree direction window should be utilized because it is more conservative than utilizing a 22.5 degree direction window. Reference 7.2.24 shows that, during the code's calculation process, ARCON96 compares the wind direction found in the hourly meteorological data to the wind direction window that contains the wind directions assumed to carry the effluent from the release point to the receptor. If the wind direction does not fall within the direction window, the X/Qs are set to zero. A smaller direction window inherently produces more zero values, which effectively lowers the final 95th percentile X/Q since ARCON96 includes these zeroes in the hourly averaging period calculations which are used in calculating the 95th percentile X/Q. Therefore, using a larger direction window results in more non-zero X/Qs in the hourly averaging periods, and thus a larger (more conservative) final 95th percentile relative concentration.

4.1.4 Atmospheric Dispersion Estimates in the Vicinity of Buildings

Reference 7.2.23 describes revisions made to the 1995 standard methodology used for estimating relative concentrations in the vicinity of buildings. The revised model later became the industry standard model, and its methodology was used to create ARCON96. The revised model includes corrections to the diffusion coefficients specifically implemented to improve model performance at low wind speeds, where meander and possibly uneven heating of building surfaces may be responsible for increased diffusion and at high wind speeds where turbulence from wakes dominates. This reference contains a section that validates the revised model through comparison of calculated relative concentrations and observed relative concentrations as illustrated in Figure 4-1. The methodology from RG 1.145 is included in this figure for comparison.

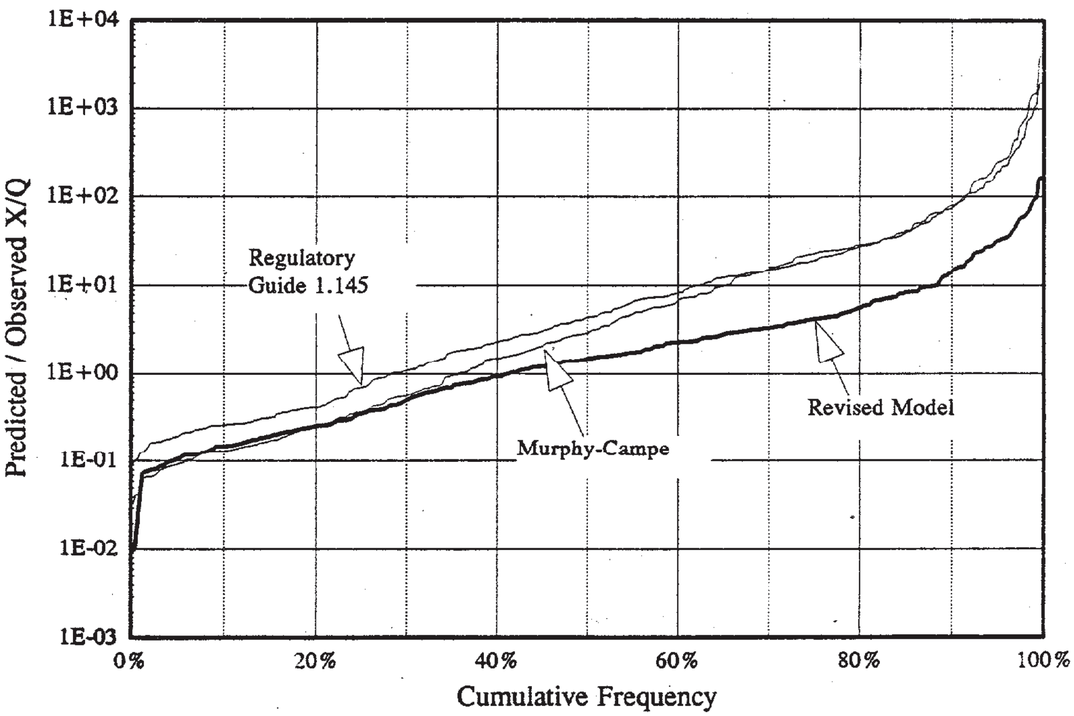

Figure 4-1. Cumulative frequency distributions of predicted to observed concentration ratios for the Murphy-Campe (RG 1.145), and revised models (Reference 7.2.23)

Figure 4-1 shows that compared with other NRC models, the revised model has less tendency to over-predict relative concentrations, especially at cumulative frequencies above 40 percent. At ratio cumulative frequencies of 95 percent and greater, as shown in Figure 4-1, RG 1.145 methodology over-predicts relative concentrations by two to three orders of magnitude and the revised model over-predicts relative concentrations by one to two orders of magnitude.

The observed relative concentrations were recorded from various experiments. The distances range from 8 to 1200 meters, meteorological condition stability classes range from extremely unstable (1) to extremely stable (7), and the wind speeds range from less than 1 m/s to greater than 10 m/s. Table 4-1 includes other relevant meteorological statistics from the data set. The emphasis on low wind speed, (where “low wind speed” is assumed to be wind speeds of less than 4 m/s in the context of this report), stable conditions is appropriate because concentrations predicted for these conditions typically provide the limiting case in evaluation of consequences of accidental releases in the vicinity of buildings. Specifically, Reference 7.2.15 shows that the ARCON96 95th percentile relative concentrations are typically associated with wind speeds of 3 to 4 m/s.

Table 4-1. Meteorological statistics from data set in Figure 4-1(Reference 7.2.23)

Description	**Number of points**	**Percentage of set**
Wind speed < 4 m/s	253	67
Stable atmospheric conditions	208	55
Low wind speed and stable atmospheric conditions	138	36
Total	379	100

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Figure 4-2. Bias in RG 1.145 model concentration predictions (Reference 7.2.23)

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Figure 4-3. Bias in ARCON96 concentration predictions (Reference 7.2.23)

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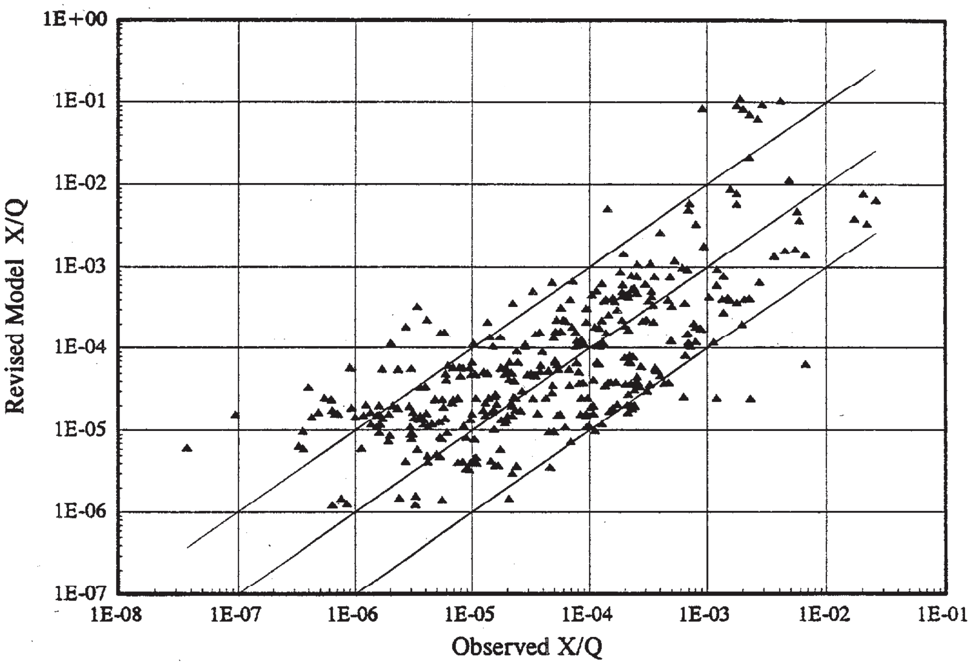

Figure 4-4. Comparison of ARCON96 concentration predictions with observed values (Reference 7.2.23)

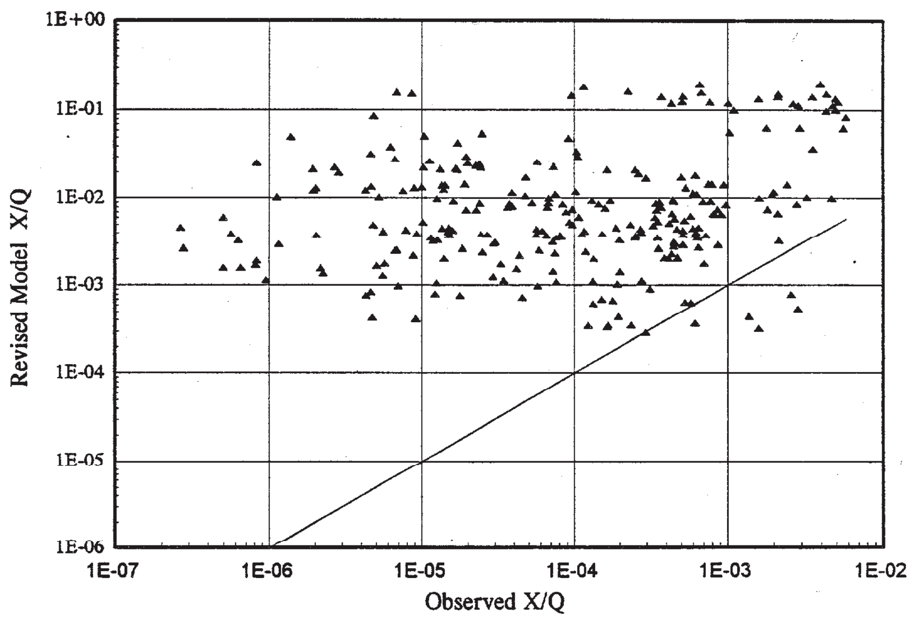

Figure 4-5. Comparison of ARCON96 concentration estimates with observed values in the building surface data set (Reference 7.2.23)

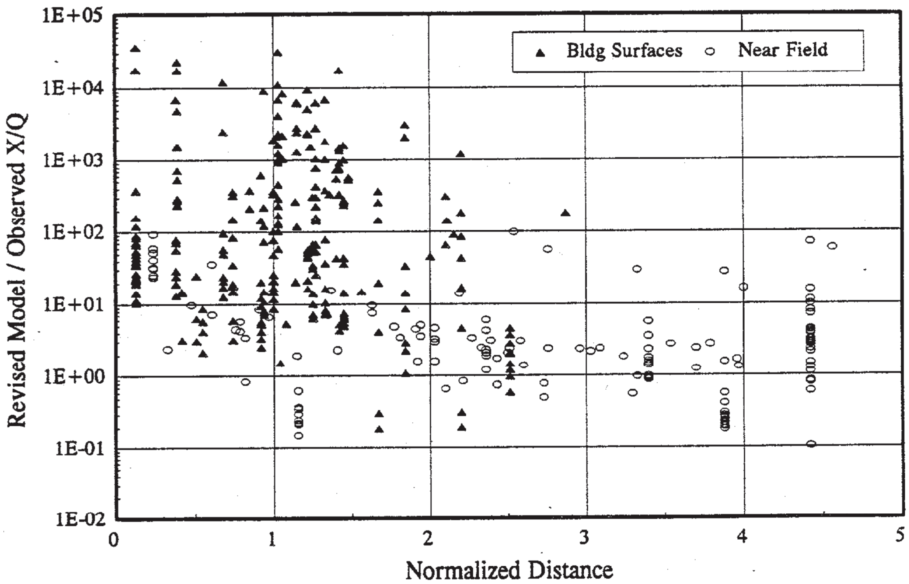

Figure 4-6. Ratios of predicted to observed concentrations for ARCON96 (Reference 7.2.23)

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Figure 4-7. Variation of low speed diffusion coefficient increments as function of distance (Reference 7.2.23)

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Figure 4-8. Variation of high speed diffusion coefficient Increments as function of distance (Reference 7.2.23)

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4.1.5 PAVAN and ARCON96 Comparison

PAVAN, which is based upon RG 1.145 methodology, is the industry-standard software for the calculation of offsite relative concentrations. However, as noted in Section 4.1, the postulated NuScale EAB and outer LPZ boundary distances are in the near vicinity of on-site buildings, and are much shorter than distances associated with a nuclear power plant of standard size. The downwind distances associated with the observed values in Figure 4-1 are in the range of 8 to 1200 meters. This range adequately encompasses the postulated range of the NuScale EAB and LPZ (approximately 80 to 400 meters).

As Figure 4-2 illustrates, the PAVAN methodology over-predicts relative concentrations at low wind speeds. Because of this over-prediction, PAVAN is not a realistic model for a plant with a small LPZ and EAB. This result is consistent with the quote from Reference 7.2.23 provided in Section 4.1 of this report. The guidance referred to in the quote from Reference 7.2.23 is the Murphy-Campe methodology, which is similar to the RG 1.145 methodology with regard to predictive ability. This correlation between RG 1.145 and the Murphy-Campe methodology is shown in Figure 4-1. RG 1.145 methodology over-predicts concentrations more frequently than the Murphy-Campe model. Since the Murphy-Campe methodology is defined as “overly conservative” in the vicinity of

buildings, per Reference 7.2.23, then the RG 1.145 methodology is also assumed to over-predict in these same conditions and is not a realistic model for a plant with a small LPZ and EAB.

ARCON96, which is based upon the aforementioned “revised model,” is the industry-standard software for the calculation of control room relative concentrations in the vicinity of buildings. As Figure 4-1 and Figure 4-3 illustrate, the revised model is more accurate than the other models at the limiting case, and its predictions provide sufficient margin.

4.1.5.1 Test Case One: Percentile Comparison

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Table 4-2. Meteorological statistics from data set in Figure 4-1 and test case

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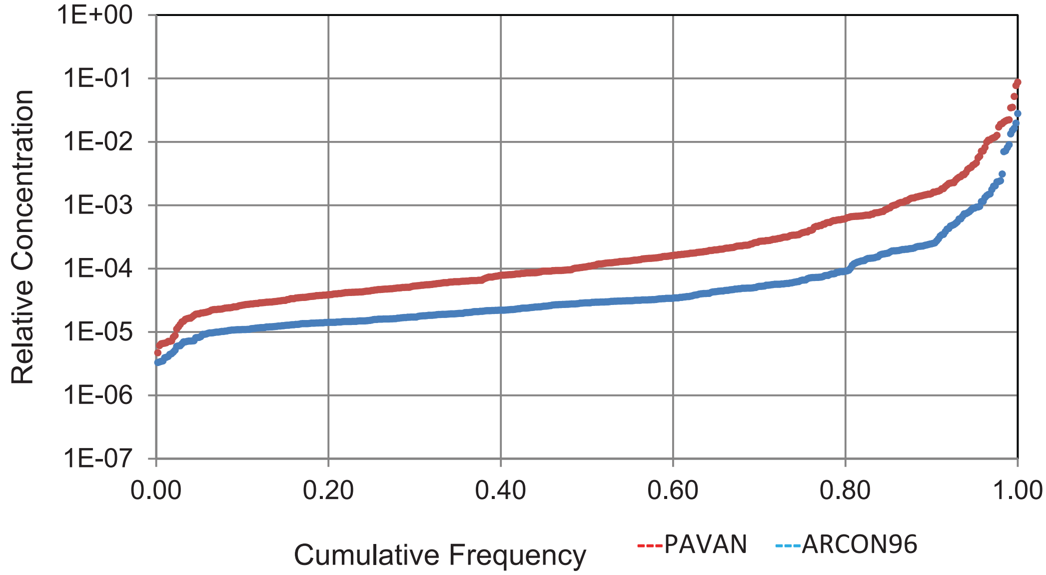

Figure 4-9. Cumulative frequency distributions of predicted concentrations for PAVAN and ARCON96 methodologies

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4.1.5.2 Test Case Two: Distance Comparison

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Figure 4-10. Ratio of PAVAN to ARCON96 versus distance (data from Figure 4-9)

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Figure 4-11. Ratio of PAVAN to ARCON96 versus distance (site data)

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Table 4-3. Site meteorological statistics

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4.1.6 Application

In order to utilize ARCON96 for offsite atmospheric dispersion calculations, the following methodology is utilized. {{

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4.2 Design Basis Source Term

The DBST is composed of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of STDBAs that are utilized as inputs in radiological consequence calculations associated with the MHA.

The DBST is assumed as the design basis case for evaluating the performance of release mitigation systems, including intact containment, as described in 10 CFR 52.47 (a)(2)(iv), and for evaluating the proposed siting of a facility. The DBST is based on a major accident, postulated for the purpose of design analyses. Events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products are addressed in the methodology.

RG 1.183 specifies that the DBST "…must represent a spectrum of credible severe accident events." Current NuScale probabilistic risk assessment (PRA) results indicate low accident sequence frequencies such that there may be no credible STDBA events for the NuScale design. However, relative risk insights are used, not to select a single risk-significant event, but to establish a range of events to be considered for the DBST calculation.

4.2.1 Definition of Source Term

A subset of the Level 1 PRA sequences is used to select the spectrum of STDBAs considered for the establishment of the DBST. These sequences are all single module internal events at full power and assume an intact containment. The methodology directs that a range of STDBA scenarios are selected that involve significant damage to the reactor core, with subsequent release of appreciable quantities of fission products.

NuScale design-specific MELCOR analyses are performed to calculate the timing and magnitude of fission product radionuclide release from failed fuel in selected core damage STDBAs. For each STDBA, key parameters such as the onset time for fission product release from the gap, duration of the gap plus early in-vessel release, and the gap plus early in-vessel release fractions for each major radionuclide group are calculated. The minimum onset time for fission product release from the gap, the minimum duration of the release, and the median value of the release fractions determined from the spectrum of STDBAs are established and used as the DBST. The DBST is the key input for calculating offsite radiological consequences as described in SRP Section 15.6.5 and 10 CFR 52.47 (a)(2)(iv).

A summary of the DBST and radiological consequence modeling approach is as follows:

- a series of equipment failures results in significant core damage
- activity released from the fuel occurs over a calculated period of time and homogenously mixes in containment atmosphere
- removal of aerosol occurs through natural processes inside the NuScale containment vessel (sedimentation)

- leakage from containment is direct to the environment, with no removal mechanisms in reactor building (scrubbing, partitioning, deposition, etc.)
- onsite and offsite radiological consequences are calculated

4.2.2 Core Damage

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4.2.2.1 Radionuclide Groups

Release fractions for nine major radionuclide groups are evaluated as part of the methodology. Table 14 in the Sandia National Lab report SAND2011-0128 (Reference 7.2.10) provides an alternative set of radionuclide groups as compared to Table 5 of RG 1.183. This alternative set of radionuclide groups represents the current approach to severe accident progression, and therefore, was included in the methodology presented in this report. No elements are added or removed from the RG 1.183 selection in the methodology. Instead, the elements are assigned to different radionuclide groups. Specifically, the alkaline earths and molybdenum groups are added and six elements are moved into different groups (Sr, Ba, Mo, Nb, Tc, and Zr). A summary of the radionuclide groups used for the methodology presented in this report are provided in Table 4-4.

Table 4-4. Radionuclide groups

Number	Name	Elements in Group
1	Noble Gases	Kr, Xe
2	Halogens	Br, I
3	Alkali Metals	Rb, Cs
4	Tellurium Group	Se, Sb, Te
5	Alkaline Earths	Sr, Ba
6	Molybdenum Group	Mo, Nb, Tc
7	Noble Metals	Ru, Rh, Pd, Co
8	Lanthanides	La, Nd, Eu, Pm, Pr, Sm, Y, Cm, Am
9	Cerium Group	Ce, Pu, Np, Zr

4.2.3 Release Timing and Magnitude

As with radionuclide groups, design-specific representative results for release timing and magnitude from severe accident evaluations are utilized for the methodology, in order to reflect current practices and appropriately model the specific event.

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4.2.4 Aerosol Transport Analysis

As discussed in detail in Section 4.3 of this report, natural deposition phenomena including sedimentation, diffusiophoresis, thermophoresis and hygroscopicity result in aerosol removal. The aerosol removal methodology described in this report utilizes the aerosol removal code STARNAUA to track these various deposition phenomena in calculating time-dependent airborne aerosol mass and removal rates. {{

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A summary of the aerosol transport and removal calculation process is described as follows:

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4.2.5 Radiological Consequence Analysis

The DBST onsite and offsite radiological consequences follow the general methodology described in Section 3.0 of this report, with event-specific radionuclide groups, release timing and fractions, and aerosol removal. The aerosol removal rate as a function of time is input into RADTRAD, from which, along with other key inputs such as atmospheric dispersion factors and isotopic inventories, the radiological consequences are calculated.

The chemical form of radioiodine released to the containment atmosphere is assumed to be 95 percent cesium iodide (CsI), 4.85 percent elemental iodine, and 0.15 percent organic iodide in accordance with Appendix A of RG 1.183. The methodology considered cesium iodide as an aerosol.

4.3 Aerosol Removal and Transport

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Table 4-5. Summary of empirical aerosol parameter ranges

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4.3.1 STARNAUA

As described in Section 3.1.6, STARNAUA models natural removal of containment aerosols by gravitational settling and diffusiophoresis, and considers the effect of hygroscopicity (growth of hygroscopic aerosols due to steam condensation on the aerosol particles) on aerosol removal. {{

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4.3.2 Sedimentation

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4.3.3 Phoretic Phenomena (Diffusiophoresis and Thermophoresis)

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4.3.4 Hygroscopicity

Hygroscopicity, or the measure of a molecule's water solubility, is an important phenomenon of interest in that it impacts aerosol particle growth and, therefore, sedimentation rate. {{

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4.3.5 Elemental Iodine Removal

Deposition of elemental iodine onto inner-containment surfaces results in its removal as a source term for release to the environment. A key assumption of this report's aerosol transport methodology is that there is no maximum limit on elemental iodine decontamination factor. The basis for this assumption is that this removal is facilitated by natural processes, as opposed to an active spray system. Additionally, as described in the following sections, the removal rate calculation methodology for this report is based on calculated time-dependent airborne aerosol mass. This methodology is in agreement with Section 3.3 of Appendix A of RG 1.183 in which it is stated that "…reduction in the removal rate is not required if the removal rate is based on the calculated time-dependent airborne aerosol mass."

4.3.6 Aerosol Resuspension and Revaporization

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4.3.7 Charge Effects on Aerosol Removal Rates

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4.3.8 Aerosol Plugging

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4.3.9 Experimental Benchmarking and Code Validation

Experimental benchmarking of STARNAUA is performed as one basis for demonstrating the validation and applicability of this modeling tool to the expected conditions of the post-accident NuScale containment. {{

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Aerosol removal models were developed in STARNAUA for the purpose of benchmarking against experimental data from the LWR Aerosol Containment Experiment (LACE) and Aerosol Behavior Code Validation and Evaluation (ABCOVE) tests. In all, benchmark models were prepared to compare calculated airborne aerosol

concentration values with measured airborne concentration values from LACE tests LA4 and LA6, and ABCOVE tests AB5 and AB7. All models incorporated the previously discussed sedimentation, diffusiophoresis, thermophoresis, and hygroscopic removal phenomena. Brief descriptions of the tests are as follows.

- LA4 – Aerosol depletion with overlapping injection periods of mixed hygroscopic and non-hygroscopic aerosol, aerosol growth by agglomeration and condensation of water vapor
- LA6 – Aerosol depletion with simultaneous injection of mixed hygroscopic and non-hygroscopic aerosol, aerosol growth by agglomeration and condensation of water vapor, rapid containment depressurization
- AB5 – Aerosol depletion with a single component hygroscopic aerosol, aerosol growth by agglomeration
- AB7 – Aerosol depletion with two hygroscopic aerosol species, aerosol growth by agglomeration

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}}2(a),(c) All benchmark models with the exception of LA4 satisfied the acceptance criteria. In the case of LA4, failure to meet the selected approval criterion is explained in light of observations made in the LA4 test report (Reference 7.2.50) regarding the expectation of generally large standard deviations for concentration measurements taken after the start of the test's vent period. {{

}}2(a),(c) Based upon these observations, STARNAUA is judged to benchmark acceptably well for test LA4 and is determined to be adequate for use in an environment in which leakage occurs such as postulated in the NuScale analysis. Experimental benchmarking results are shown in Table 4-6 and Figure 4-12 through Figure 4-15.

Table 4-6. STARNAUA experimental benchmarking results

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Figure 4-12. Calculated and measured suspended aerosol concentrations in Test LA 4

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Figure 4-13. Calculated and measured suspended aerosol concentrations in Test LA 6

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Figure 4-14. Calculated and measured suspended aerosol concentrations in Test AB 5

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Figure 4-15. Calculated and measured suspended aerosol concentrations in Test AB 7

4.3.10 Benchmarking to MAEROS

MAEROS, a component of MELCOR, may be used to simulate the evolution of an aerosol based on the particle mass. Physical phenomena included in the calculations are Brownian coagulation, gravitational coagulation, diffusion, thermophoresis, and deposition. Models for diffusiophoresis and hygroscopicity are not included in the MAEROS code.

As described in Reference 7.2.65, the general numerical approach of MAEROS uses sections (bins) defined by particle mass instead of particle dimensions. For N sections, MAEROS calculates a set of 2N(N+2) sectional coefficients. Each coefficient is a rate constant for a different transport mechanism. These coefficients, for a given containment, often depend only on gas temperature and pressure. Given this condition, the user can specify upper and lower temperature and pressure limits. These limits give four sectional coefficients corresponding to the four combinations of temperature and pressure limits. MAEROS is able to linearly interpolate temperature and pressure values that differ from the limits and perform calculations with aerosol components, where a component is a physical constituent of the aerosol.

As described in Reference 7.2.43, MAEROS was originally developed in part to eliminate numerical diffusion that occurred in early aerosol codes through the use of a “moving boundary” numerical scheme for bin sizing. {{

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Table 4-7. Test geometry

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Figure 4-16. Benchmark Case 1 CsOH suspended concentration

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Figure 4-17. Benchmark Case 2 CsOH suspended concentration

Overall, the benchmark resulted in close agreement between codes for the two evaluated cases, with mean STARNAUA-to-MAEROS ratio of {{ }}2(a),(c) for Case 1 and {{ }}2(a),(c) for Case 2. This results in an additional independent means of demonstrating the ability of STARNAUA for prediction of aerosol coagulation and diffusion.

4.3.11 Application

The description of the methodology to use STARNAUA for the DBST is provided in Section 4.2.4 of this report. This methodology assumes NuScale-specific release magnitude and timing and thermal-hydraulic conditions calculated by MELCOR. General methods for calculating inputs for STARNAUA are described in this section.

The aerosol species released into the containment are classified into the chemical groups defined in Table 4-4 {{

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Table 4-8. Radionuclide group molecular mass multipliers

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4.4 DBST Sensitivity Analysis

4.4.1 General Sensitivity Analysis Methodology

General sensitivity analysis methodologies are used throughout Section 4.4 and are briefly mentioned here for convenience.

A statistically based nonparametric input sampling process is implemented through automated software tools. Input parameters are randomly varied across their pertinent range of values within the input deck and run as a single sample whose output file gives a value. Scatter plots are created to visually illustrate the sensitivity of the system to each individual input and sensitivity metrics are calculated to assist in identifying trends seen in the scatter plots.

The partial rank correlation coefficient (PRCC) value is also calculated. A positive PRCC value means that the effect of the input on the output is the same, i.e., an increase of the value of the input leads to an increase of the value of the output. A negative PRCC value means the effect is the opposite, i.e., an increase in the input leads to a decrease in the output.

The primary indicator of importance is the incremental R^2 from the quadratic regression model. An input is not sufficiently important if it has an incremental R^2 less than 0.02. A high incremental R^2 (close to 1.0) indicates that an input is highly influential on the evaluated system output.

4.4.2 Application to DBST

General sensitivity analysis methodologies are applied to the DBST calculation, including the aerosol modeling component, in order to provide a quantitative evaluation of the impact of aerosol modeling on the key output of the DBST calculation, which is the radiological consequence.

The discussion on the results of the sensitivity analysis is focused on those inputs which are shown to be most influential on control room dose, LPZ dose, and aerosol removal rate. EAB dose is not included as it focuses only on the limiting two hour window. {{

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Table 4-9 presents the inputs sampled for the example sensitivity analysis utilizing uniform distributions. Minimum and maximum values of empirically observed aerosol parameters from core damage and aerosol specific experiments performed for LWRs are defined in Table 4-5. {{

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Table 4-9. Summary of sampled input assumed for sensitivity analysis

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Observations of the quadratic regression, PRCC, and adjusted R^2 values for the example sensitivity analysis are described in the following paragraphs. In all cases the adjusted R^2 indicated fair to good performance with quadratic values ranging approximately from 0.7 to 0.9, depending on the number and type of inputs for a case. Considering the fair to good performance of the adjusted R^2, non-linear effects with respect to sensitivity analysis are ruled out, and the resulting sensitivity analysis is taken to be reasonable. A minimum of 1,000 samples were utilized in the analysis and taken to be appropriate based on acceptable adjusted R^2 values.

The quadratic regression criteria consistently aligned across all cases and with the PRCC rankings. {{

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As noted previously, one use of this analysis is to check for the possibility of a non-linear system with corresponding feedback effects. It is important to check for a non-linear system as parameter importance and bias directions are not consistent in such systems. Thus, in the case of a non-linear system, conditional bias directions would have to be established in order to ensure a robust deterministic methodology.

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Table 4-10 and Table 4-11 present the key parameters (as defined by the quadratic regression criteria) for control room and LPZ dose results, assuming constant aerosol inputs. This case is used to set a baseline for the example sensitivity analysis and to provide context with respect to aerosol inputs. The resulting relative importance and bias directions are in alignment with expectations.

Table 4-10. Key control room dose input rankings and bias directions

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Table 4-11. Key LPZ dose input rankings and bias directions

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Figure 4-18 and Figure 4-19 present example control room and LPZ dose results assuming constant aerosol inputs in trace plot format. {{

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Figure 4-18. Example control room dose trace plot (constant aerosol)

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Figure 4-19. Example LPZ dose trace plot (constant aerosol)

Figure 4-20 and Figure 4-21 present bar charts of the calculated partial rank correlation coefficients, an indicator of the importance of an input, for control room and LPZ dose results assuming constant aerosol inputs. This information is the same information as presented in Table 4-10 and Table 4-11 in a visual format.

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Figure 4-20. Example control room dose sensitivity rankings (constant aerosol)

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Figure 4-21. Example LPZ dose sensitivity rankings (constant aerosol)

Aerosol-specific inputs are added to the sensitivity analysis, with results presented in Table 4-12, Table 4-13, Figure 4-22, and Figure 4-23. The example LPZ dose results are described in the following paragraphs, which are representative of both control room and EAB radiological consequences.

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Table 4-12. Key aerosol inputs for LPZ dose rankings and bias directions

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Table 4-13. Key aerosol concentration input rankings and bias directions

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Figure 4-22. Low population zone dose trace plot

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Figure 4-23. Aerosol concentration trace plot

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}}2(a),(c) Table 4-14 provides the sensitivity analysis established bias directions for parameters to maximize dose and minimize aerosol removal.

Table 4-14. Direction of bias to maximize dose and minimize aerosol removal

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4.4.3 Sensitivity Analysis Conclusions

Results from the example input sensitivity analysis performed for the DBST are used in the methodology to determine the appropriate biasing direction for an input in order to yield a conservative solution. In addition, this example analysis may be utilized as a supporting determination of the linear nature with respect to sensitivity analysis (i.e., consistent biasing directions) of the system modeled. A high-level summary and

conclusions of the example sensitivity analysis with respect to radiological consequences and aerosol removal is provided:

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4.5 Post-Accident pH_T

This section presents the methodologies utilized for evaluating post-accident pH_T in coolant water following a significant core damage event such as the DBST. The symbol "pH_T" is used to emphasize dependence on temperature. This term is synonymous with the traditional symbol pH, which is typically calculated at room temperature. The pH_T is used for calculating the extent of iodine re-evolution inside containment. During the postulated DBST event, additional acids and bases are expected to enter the coolant and cause a change in pH_T. The overall pH_T of the coolant is expected to be modeled over a time period of 30 days. Though it's not expected, a design that utilizes this methodology could contain halogenated cable insulation in containment. Therefore, hydrochloric acid is included in the methodology. A summary of the parameters that influence pH_T are as follows:

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The pH_T of a system is dependent on the concentration of the hydronium (H_3O^+) in the aqueous system. The hydronium concentration, often simplified as H^+, is controlled by the self-dissociation of water and the presence of any acids and/or bases. The pH_T of an aqueous system is calculated as

$$\mathrm{pH_T} = -\mathrm{Log}([\mathrm{H^+}]) \quad \text{Eq 4-25}$$

The hydronium concentration H^+ can be calculated from a system of equations that includes: the dissociation of acids/bases, mass balance of the chemical species, and charge balance of all ions in the system. Concentrations used in this calculation are in terms of molality. Molality is defined as

$$\frac{moles\ of\ solute}{mass\ (kg)\ of\ solvent} = molality\ of\ solute \quad \text{Eq 4-26}$$

4.5.1 Dissociation Equation

The dissociation equation describes to what extent each chemical species dissociates (separates) into ionic form. This dissociation of a generic acid or base will form an equilibrium described by

$$\mathrm{HA} \leftrightarrow \mathrm{H^+} + \mathrm{A^-} \quad \text{Eq 4-27}$$

$$\mathrm{BOH} \leftrightarrow \mathrm{B^+} + \mathrm{OH^-} \quad \text{Eq 4-28}$$

Water is unique in that it undergoes self-ionization, where a water molecule dissociates into a hydrogen and hydroxide ion.

$$\mathrm{H_2O} \leftrightarrow \mathrm{H^+} + \mathrm{OH^-} \quad \text{Eq 4-29}$$

The rate at which these reactions occur at equilibrium is governed by the dissociation constant *K* which is defined as the ratio of concentrations of products to reactants. The dissociation constant for water, a generic acid, and a generic base are defined as

$$\mathrm{K_{H_2O}} = [\mathrm{H^+}] * [\mathrm{OH^-}] \quad \text{Eq 4-30}$$

$$\mathrm{K_{HA}} = \frac{[\mathrm{H^+}] * [\mathrm{A^-}]}{[\mathrm{HA}]} \quad \text{Eq 4-31}$$

$$K_{BOH} = \frac{[B^+] * [OH^-]}{[BOH]} \quad \text{Eq 4-32}$$

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Table 4-15. Dissociation constants of assumed acids and bases

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4.5.2 Mass Balance Equation

The mass balance equation conserves the total mass of an element or compound found in the coolant. The equations for a generic acid and base are defined as

$$[A]_{Total} = [HA] + [A^-] \quad \text{Eq 4-33}$$

$$[B]_{Total} = [BOH] + [B^+] \quad \text{Eq 4-34}$$

4.5.3 Charge Balance Equation

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4.5.4 Final Charge Balance Equation

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4.5.5 Concentration of Ionic Species

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4.5.5.1 Equations for Concentration

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Table 4-16. Concentration equations of included chemical species

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Table 4-17. Concentration equations of boric acid ionic species

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4.5.5.2 $[B(OH)_3]$ Solution

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4.5.6 Iodine Re-evolution

An estimate for iodine re-evolution was made using Figure 3-1 of NUREG/CR-5950 (Reference 7.2.51), shown as Figure 4-24 in this report. This data was used as the basis for an acceptance criteria of greater than or equal to seven in Reference 7.2.2. According to this figure, less than one percent of aqueous iodine is converted into molecular I_2 at a pH_T of 6.0. The derived methodology is performed at 25 degrees Celsius to match available experiment data; thus no temperature dependence is included in the model.

The methodology of this report assumes that for pH_T values of 6.0 or greater, the negligible amount of iodine re-evolution that could occur between pH_T values of 6.0 and 7.0 does not need to be explicitly included in the dose analysis calculation. This position simplifies the analysis without an impact to the conservatism of the calculated dose results.

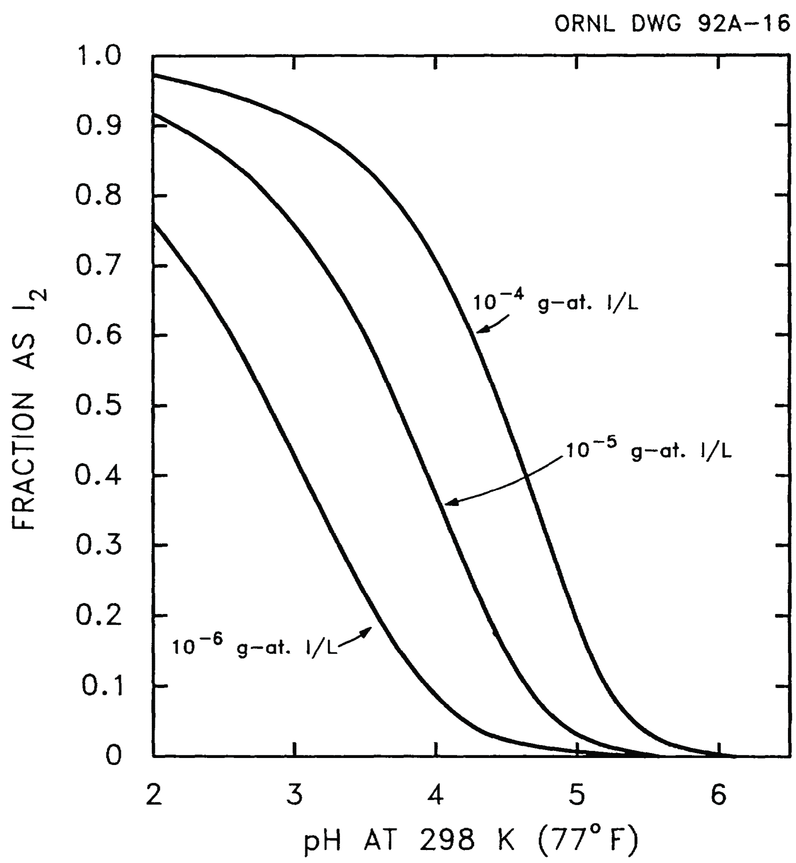

Figure 4-24. Iodine re-evolution versus pH (Reference 7.2.51)

5.0 Example Calculation Results

Example calculation analyses and results are presented in this section to demonstrate the application of the methodology described in this report. These results are for illustrative purposes. NuScale plans to provide the final design values in the design certification application. Examples are provided in this section for offsite and onsite atmospheric dispersion factors, severe accident event selection, example severe accident analysis, containment aerosol removal, Category 1 and 2 radiological consequences, and post-accident pH_T.

5.1 Atmospheric Dispersion Factors

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Figure 5-1. Map of each surface data site in the selected EPA dataset

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5.1.1 Offsite Dispersion Factors

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Figure 5-2. Markup of site layout with analytical offsite distances overlaid

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Table 5-1. Time-interval relative concentrations for selected site

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Figure 5-3. Histogram of calculation results at 33 meters

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Figure 5-4. Histogram of calculation results at 122 meters

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Table 5-2. Ratio of selected relative concentration to true 90th percentile

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Table 5-3. Selected meteorological data

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Table 5-4. Example offsite atmospheric relative concentration (X/Q) values

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5.1.2 Control Room and Technical Support Center Dispersion Factors

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Table 5-5. Example control room atmospheric dispersion factors

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5.2 Category 1 Events

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Table 5-6. Example dose results for Category 1 events

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5.3 Example DBST Selection Process

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Table 5-7. Spectrum of example STDBAs cases considered for creation of DBST

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5.4 Example Severe Accident Analysis

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Table 5-8. Example severe accident timeline of notable events

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Figure 5-5. Example STDBA No. 2 short-term RPV and CNV pressures

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Figure 5-6. Example STDBA No. 2 long-term RPV and CNV pressures

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Figure 5-7. Example STDBA No. 2 RPV and CNV collapsed liquid levels

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Figure 5-8. Example STDBA No. 2 distribution of coolant

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Figure 5-9. Example STDBA No. 2 peak cladding temperature

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Figure 5-10. Example STDBA No. 2 representative containment temperatures

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Figure 5-11. Example STDBA No. 2 release fractions from fuel

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Figure 5-12. Example STDBA No. 2 release fractions into containment

5.5 Representative Severe Accident Results

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Table 5-9. Comparison of release timing and magnitudes of example STDBAs

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5.6 Example Containment Aerosol Transport and Removal

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Table 5-10. Example accident scenarios for aerosol simulation

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Figure 5-13. Baseline case aerosol concentration and removal

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Figure 5-14. Baseline case aerosol average radius

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Table 5-11. Summary of key parameters from all cases

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Figure 5-15. Comparison of aerosol concentration for all example cases versus time

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Figure 5-16. Comparison of aerosol removal rate for example cases versus time

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Table 5-12. Summary of example aerosol removal results

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5.7 Example DBST Radiological Consequences

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Table 5-13. Summary of example RADTRAD case results

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5.8 Post-Accident pH_T

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Table 5-14. Summary of example pH_T results for calculations performed at 25 degrees C

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Table 5-15. Summary of example results for baseline calculation with increasing temperatures

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Figure 5-17. The pH_T of the coolant over a 30 day time period

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Figure 5-18. Effect of elevated temperature on pH_T

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Figure 5-19. Sensitivity of pH_T to boron concentration

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Figure 5-20. Sensitivity of pH_T to cesium hydroxide

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Figure 5-21. Sensitivity of pH_T to nitric acid and hydrochloric acid

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Figure 5-22. Sensitivity of pH_T to the mass of liquid coolant in containment

6.0 Summary and Conclusions

A methodology for developing accident source terms and performing the corresponding radiological consequence analyses was presented in this report. The methodology was shown to be conservative by providing example results from sensitivity analyses. Key unique features of the methodology presented in this report are the use of ARCON96 for offsite atmospheric dispersion factors, the development of a DBST to meet the intent of 10 CFR 52.47 (a)(2)(iv), the utilization of STARNAUA containment aerosol transport code in the range of NuScale's expected post-accident containment conditions, and evaluation of post-accident pH_T.

ARCON96 was found to be suitably conservative for NuScale's intended use of the code as a substitute for PAVAN for offsite atmospheric dispersion factor calculations. As presented in this report, the ARCON96 methodology has less of a tendency to over-predict concentrations, while still providing predictions that are sufficiently conservative.

The STARNAUA containment aerosol transport and removal code was benchmarked against experimental data and was shown to be appropriate for modeling aerosol removal in the DBST analysis associated with the post-accident containment conditions. Consistent with RG 1.183, the assumption that no elemental iodine decontamination factor limit should be applied to natural aerosol removal phenomenon was utilized for modeling removal in the containment vessel. Through example sensitivity analysis on the modeling parameters utilized as input to the STARNAUA code, {{

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This insight provides confidence that the aerosol modeling methodology is robust and the inputs utilized for the DBST analysis are conservative in the NuScale design certification application.

Example calculations were provided in order to demonstrate applicability of the methodology.

6.1 Criteria for Establishing Applicability of Methodologies

The generalized methodologies presented in this topical report are based upon numerous modeling assumptions. For completeness, the following set of criteria for establishing the applicability of these methodologies is provided. The design certification application or combined operating license application that utilizes the methodology of this topical report must satisfy these criteria in order to establish applicability. Any deviations to these criteria must be explicitly defined and justified as part of the application that references this topical report.

6.1.1 Criteria for Atmospheric Dispersion Factors

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6.1.2 Criteria for Core Radionuclide Inventory

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6.1.3 Criteria for Control Room Modeling

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7.0 References

7.1 Source Documents

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Enclosure 3:

Affidavit, AF-0416-48655

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 process and method by which NuScale develops its Accident Source Term Methodology.

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

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

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

(4) The information sought to be withheld is in the enclosed report titled "Accident Source Term Methodology." 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 April 8, 2016

Thomas A. Bergman