February 26, 2020 Docket No. PROJ0769

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

SUBJECT: NuScale Power, LLC Submittal of the Approved Version of NuScale Topical Report "Accident Source Term Methodology," TR-0915-17565, Revision 3

REFERENCE: NRC Letter to NuScale, "Final Safety Evaluation for NuScale Power, LLC, Licensing Topical Report: TR-0915-17565, Revision 3, 'Accident Source Term Methodology,'" dated January 30, 2020 (ML20027A105)

By referenced letter dated January 30, 2020, the Nuclear Regulatory Commission (NRC) issued a final safety evaluation report documenting the NRC Staff conclusion that the NuScale topical report TR-0915-17565, "Accident Source Term Methodology," Revision 3, as updated by the July 31, 2019 letter, is acceptable for referencing in licensing applications for the NuScale small modular reactor design. The referenced NRC letter requested that NuScale publish the approved version of TR-0915-17565, within three months of receipt of the letter.

NuScale submitted TR-0915-17656 "Accident Source Term Methodology," Revision 3 on April 21, 2019 (ML19112A172). At the direction of the NRC, NuScale submitted updates to Revision 3 in a letter dated July 31, 2019 (ML19212A802). The updates, together with Revision 3, became known as Revision 4 to the "Accident Source Term Methodology" topical report. Therefore, Revision 4, not Revision 3, is provided as the approved version of the topical report in this letter.

Accordingly, Enclosure 1 to this letter provides the approved version of TR-0915-17565-P-A, Revision 4. This enclosure includes the January 30, 2020 NRC letter and its final safety evaluation report, the NuScale response to NRC requests for additional information, and documentation of the final topical report submittal, Revision 3.

Enclosure 1 contains proprietary information. NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of the approved topical report package.

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

If you have any questions, please feel free to contact Matthew Presson at 541-452-7531 or at mpresson@nuscalepower.com if you have any questions.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12
Michael Dudek, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Topical Report, TR-0915-17565-P-A, "Accident Source Term Methodology," Revision 4, proprietary version
Enclosure 2: NuScale Topical Report, TR-0915-17565-NP-A, "Accident Source Term Methodology," Revision 4 nonproprietary version
Enclosure 3: Affidavit of Zackary W. Rad, AF-0220-68979

Enclosure 1:

NuScale Topical Report, TR-0915-17565-P-A, "Accident Source Term Methodology," Revision 4, proprietary version

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

Enclosure 2:

NuScale Topical Report, TR-0915-17565-NP-A, “Accident Source Term Methodology,” Revision 4, nonproprietary version

NuScale Power, LLC
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www.nuscalepower.com

Contents

Section A

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

January 30, 2020

Mr. Zackary W. Rad
Director, Regulatory Affairs
1100 Circle Boulevard, Suite 200
Corvallis, OR 97330

SUBJECT: FINAL SAFETY EVALUATION FOR NUSCALE POWER, LLC, LICENSING TOPICAL REPORT: TR-0915-17565, REVISION 3, "ACCIDENT SOURCE TERM METHODOLOGY"

By letter dated April 23, 2019, NuScale Power, LLC (NuScale) submitted licensing topical report (TR) TR-0915-17565, Revision 3, "Accident Source Term Methodology," dated April 21, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19112A172—nonproprietary version), for review and approval by the U.S. Nuclear Regulatory Commission (NRC) in support of the application for the design certification of NuScale's small modular reactor (SMR). By letter dated July 31, 2019 (ADAMS Accession No. ML19212A801), NuScale submitted changes to Revision 3 of the TR in response to the staff request for additional information.

The NRC staff has found that TR-0915-17565, Revision 3, as updated by the July 31, 2019, letter acceptable for referencing in licensing applications for the NuScale SMR design to the extent specified and under the conditions and limitations delineated in the enclosed safety evaluation report (SER). The SER defines the basis for acceptance of the TR.

In accordance with the guidance provided on the NRC's TR Web site (http://www.nrc.gov/about-nrc/regulatory/licensing/topical-reports.html), we request that NuScale publish an accepted version of this TR within three months of receipt of this letter. The accepted version shall incorporate this letter and the enclosed safety evaluation between the title page and the abstract. It must be well indexed such that information is readily located. Also, it must contain in its appendices historical review information, such as questions and accepted responses, and original report pages that were replaced. The accepted version shall include an "-A" (designated accepted) following the report identification symbol.

If the NRC's criteria or regulations change so that its conclusion in this letter (that the TR is acceptable) is invalidated, NuScale and/or the applicant referencing the TR will be expected to revise and resubmit its respective documentation or submit justification for the continued applicability of the TR without revision of the respective documentation.

If you have any questions or comments concerning this matter, I can be reached at (301) 415-8013 or via e-mail address at Getachew.Tesfaye@nrc.gov.

Sincerely,

/RA/

Getachew Tesfaye, Senior Project Manager
New Reactor Licensing Branch
Division of New and Renewed Licenses
Office of Nuclear Reactor Regulation

Docket No. 52-048

Enclosure:
As stated

cc: w/o encl.: DC NuScale Power, LLC Listserv

SUBJECT: NUSCALE POWER, LLC, ADVANCE SAFETY EVALUATION REPORT FOR LICENSING TOPICAL REPORT: TR-0915-17565, REVISION 3, "ACCIDENT SOURCE TERM METHODOLOGY"
DATE: JANUARY 30, 2020

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ADAMS Accession No.: ML20027A105 *via email NRR-043

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U.S. NUCLEAR REGULATORY COMMISSION

SAFETY EVALUATION FOR NUSCALE POWER, LLC,

TOPICAL REPORT, TR-0915-17565,

"ACCIDENT SOURCE TERM METHODOLOGY"

1.0 Introduction

By letter dated April 21, 2019, NuScale Power, LLC (NuScale) submitted licensing topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," (Reference 1, Agencywide Documents Access and Management System (ADAMS) Accession No. ML19112A172—nonproprietary version), for review and approval by the U.S. Nuclear Regulatory Commission (NRC). The subject topical report describes a general methodology for developing accident source terms and performing the corresponding design-basis accident (DBA) and other required accident radiological consequence analyses to be referenced by the NuScale small modular reactor (SMR) design certification application (DCA), Part 2, final safety analysis report (FSAR), and by other applications that reference the NuScale SMR design. Portions of the topical report are marked as NuScale proprietary information.

SECY-19-0079, "Staff Approach to Evaluate Accident Source Terms for the NuScale Power Design Certification Application," dated August 16, 2019 (ADAMS Accession No. ML19107A455), is an information paper sent to the Commission. The paper describes the regulatory and technical issues raised by unique aspects of NuScale's proposed methodology and the staff's approach to reviewing the subject topical report.

This safety evaluation report (SER) is divided into seven sections: Section 1 is the introduction; Section 2 summarizes the information presented in the topical report; Section 3 presents a summary of applicable regulatory criteria and guidance; Section 4 contains the technical evaluation of NuScale's request for approval of the proposed accident source term methodology, including use of the ARCON96 methodology for the calculation of offsite atmospheric dispersion factors; Section 5 presents the conclusions of this review; Section 6 contains the restrictions and limitations on use of the topical report methodology; and Section 7 lists the references.

2.0 Summary of Application

The NuScale accident source term methodology topical report TR-0915-17565, Revision 3, describes assumptions and methodologies, including computer codes, used to develop accident source terms and calculate radiological consequences. It is intended for use in showing compliance with the following:

- siting and safety analysis requirements in Title 10 of the *Code of Federal Regulations* (10 CFR) 52.47, "Contents of Applications; Technical Information," for design certification (DC)
- control room habitability requirements in 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," Appendix A, "General Design Criteria for Nuclear

Power Plants," General Design Criterion (GDC) 19, "Control Room," and 10 CFR 50.34(f)(2)(xxviii)

- technical support center (TSC) habitability requirements in 10 CFR 50.47(b)(8) and (b)(11) and 10 CFR Part 50, Appendix E, "Emergency Planning and Preparedness for Production and Utilization Facilities," paragraph IV.E.8

The topical report also provides methods for determining DBA radiation sources for use in the evaluation of environmental qualification of equipment in accordance with 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants." NuScale submitted the topical report to seek approval for the methodology for establishing the accident source terms for the NuScale SMR design that meet the requirements in 10 CFR 52.47(a)(2)(iv) in support of the review of the NuScale SMR DCA. This topical report is designed to support certification of the NuScale SMR design and any subsequent application that references the NuScale SMR design, such as a combined license (COL) application.

As stated in 10 CFR 50.2, "Definitions," an accident source term "refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release." The topical report develops source terms for deterministic accidents for the NuScale SMR design that are similar to those used in safety and siting assessment for large light-water reactors (LWRs), as described in Chapter 15, "Transient and Accident Analysis," of NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition" (the SRP). The DBAs described in the topical report are the main steamline break (MSLB) outside containment, rod ejection accident (REA), fuel handling accident (FHA), steam generator tube failure (SGTF), and the failure of small lines carrying primary coolant outside containment. The topical report also describes an iodine spike design-basis source term (DBST)[1], which is a surrogate accident to bound potential accidents with release of the reactor coolant into the containment vessel. In addition, the topical report provides source term and accident assessment methodology for a core damage event (CDE) in which significant core damage is assumed to occur in accordance with the description of the postulated accident fission product release in Footnote 3 to 10 CFR 52.47(a)(2)(iv).

For large LWRs, the accident associated with the siting and safety analysis regulatory requirements with respect to radiological consequences has historically been a postulated loss-of-coolant accident (LOCA), in which a break in the reactor coolant system (RCS) piping results in the inability of the emergency systems to maintain core cooling with subsequent damage to the reactor core, without damage to the reactor vessel itself and with the containment remaining intact. In general, currently operating power reactors were originally licensed by using the LOCA dose analysis source term described in Atomic Energy Commission Technical Information Document TID-14844, "Calculation of Distance Factors for Power and Test Reactor Sites," dated March 23, 1962 (ADAMS Accession No. ML021720780), which is also listed as a reference in 10 CFR 100.11, "Determination of Exclusion Area, Low Population Zone, and Population Center Distance," for the siting requirements for power reactors licensed before January 10, 1997. In 1995, the NRC published NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Reference 2), which described revised accident source terms for LWRs. Regulatory Guide (RG) 1.183, "Alternative Radiological Source Terms for

[1] As discussed in Section 4.1, the applicable NRC regulations do not require classification of source terms as "design basis" or "beyond-design-basis" to demonstrate compliance with the requirements. Therefore, the staff has determined the classification of a source term as "design-basis" or "beyond design-basis" for the NuScale design is not material to the staff's findings under these regulations.

Evaluating Design Basis Accidents at Nuclear Power Reactors" (Reference 3), provides guidance on acceptable use of alternative source terms based on NUREG-1465 in DBA radiological consequence analyses in licensing actions for power reactors. The DBA LOCA source terms in TID-14844 and RG 1.183 are not intended to reflect a specific LOCA scenario, but each is intended to represent a conservative surrogate accident based on a spectrum of break sizes up through the double-ended guillotine break of the largest RCS piping. The radiological consequence analysis of this accident is intended to evaluate the performance of the containment and release mitigation systems and to evaluate the proposed siting of the facility.

The NuScale design does not include large RCS piping; therefore, the accident scenario that would result in a fission product release to containment consistent with the regulatory requirements would not be the same as for the large LWR LOCA. Instead, to address the regulatory requirements, the NuScale topical report proposes a methodology to develop a core damage source term (CDST) based on several severe accident scenarios that result in core damage, taken from the design-specific probabilistic risk assessment (PRA). This CDST is the surrogate radiological source term for a CDE.

NuScale requested NRC approval of the following specific portions of the topical report:

(1) Treatment of the CDE, postulated as a major accident for purposes of site analysis pursuant to Footnote 3 of 10 CFR 52.47(a)(2)(iv), is an appropriate beyond-design-basis event for the NuScale design.

(2) The ARCON96 methodology is appropriate for the calculation of offsite atmospheric dispersion factors

(3) [[
]]

(4) Release timing values associated with the surrogate accident scenario with the minimum time to core damage are taken as the CDST release timing values.

(5) Representative (median) release fractions from fuel into containment from the spectrum of surrogate accident scenarios are taken as the CDST release fractions.

(6) Use of radionuclide groups from the Sandia National Laboratories report SAND2011-0128, "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX [Mixed Oxide] Fuel," issued January 2011 (Reference 4), for the CDST is appropriate.

(7) The STARNAUA aerosol transport and removal software program is appropriate for modeling natural removal of containment aerosols for the NuScale design.

(8) Utilizing thermal-hydraulic data associated with the surrogate accident scenario with the minimum time to core damage is appropriate for use in STARNAUA.

(9) No maximum limit on the iodine decontamination factor for natural removal of containment aerosols is appropriate.

(10) [[
]]

(11) Use of the iodine spiking assumptions of RG 1.183 is appropriate.

(12) Use of the iodine decontamination factor assumptions of RG 1.183 for the FHA is appropriate.

(13) For accident analysis, it is appropriate to neglect the small secondary-side volume that could contain activity from primary to secondary leakage for the NuScale design.

(14) For pH_T values of 6.0 or greater, the amount of iodine re-evolution that could occur between pH_T values of 6.0 and 7.0 is negligible and not included in the dose calculation.

(15) Containment shine of the radiation in the containment airspace through the containment vessel, reactor pool water, and then through the reactor building walls or ceiling to the environment is negligible for the NuScale design.

Section 3 of the topical report presents an overview of the proposed methodology to provide source terms for evaluating the radiological consequences of accidents. Section 4 provides the methodology that is unique to NuScale. Section 5 presents example calculations to aid in understanding the methodology described in Sections 3 and 4; therefore, the staff did not evaluate Section 5 of the topical report for approval. Section 6 presents the report's conclusions.

3.0 Regulatory Basis

The regulations in 10 CFR 52.47(a)(2)(iv) require that an application for a DC include an FSAR that provides a description and safety assessment of the plant design features intended to mitigate the radiological consequences of accidents. The safety assessment analyses are intended, in part, to show compliance with the following:

- radiological consequence evaluation factors in 10 CFR 52.47(a)(2)(iv)(A) and 10 CFR 52.47(a)(2)(iv)(B) for offsite doses at the exclusion area boundary (EAB) and outer boundary of the low-population zone (LPZ),
- the control room radiological habitability requirements in 10 CFR Part 50, Appendix A, GDC 19, and 10 CFR 50.34(f)(2)(xxviii), and
- the radiological habitability requirements for the TSC in 10 CFR 50.47(b)(8) and (b)(11) and paragraph IV.E.8 of Appendix E to 10 CFR Part 50.

In addition, 10 CFR 52.47(a)(1) requires a DC applicant to provide "the site parameters postulated for the design, and an analysis and evaluation of the design in terms of those site parameters." Site parameters are the postulated physical, environmental, and demographic features of an assumed site specified in a DCA. For the assessment of the radiological consequences of accidents, a DCA FSAR contains site parameters related to accident release atmospheric dispersion factor (χ/Q) values for the EAB, LPZ, control room, and TSC.

As described in 10 CFR 52.47(a)(2)(iv), the FSAR assessment of the plant evaluates:

> The safety features that are to be engineered into the facility and those barriers that must be breached as a result of an accident before a release of radioactive material to the environment can occur. Special attention must be directed to plant design features intended to mitigate the radiological consequences of accidents. In performing this assessment, an applicant shall assume a fission product release from the core into the containment assuming that the facility is operated at the ultimate power level contemplated. The applicant shall perform an evaluation and analysis of the postulated fission product release, using the expected demonstrable containment leak rate and any fission product cleanup systems intended to mitigate the consequences of the accidents, together with

applicable postulated site parameters, including site meteorology, to evaluate the offsite radiological consequences.

Footnote 3 to the regulation describes the fission product release for this assessment:

> 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.

The regulation at 10 CFR 52.79(d)(1) requires a COL application referencing a DC to provide information sufficient to demonstrate that the characteristics of the site fall within the site parameters specified in the DC. Site characteristics (the actual physical, environmental, and demographic features of a site) are specified in a site safety analysis report for an early site permit or in an FSAR for a COL. An early site permit application will specify site characteristics related to accident release χ/Q values at the EAB and LPZ. A COL application that references a DC typically contains site characteristics related to accident release χ/Q values at the EAB, LPZ, control room, and TSC locations for comparison against the corresponding DC site parameters.

The radiological consequences of DBAs are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale Design-Specific Review Standard (DSRS) 15.0.3, "Design Basis Accident Radiological Consequence Analyses for NuScale SMR Design," issued July 2016 (ADAMS Accession No. ML15355A341).

Accident source terms are also used to develop radiation sources for other required evaluations. The regulation at 10 CFR 50.49(e)(4) requires environmental qualification of safety-related structures, systems, and components to address a radiation environment based on the "most severe design basis accident during or following which the equipment is required to remain functional." Requirements related to Three Mile Island that use core damage source terms for evaluation of shielding for vital area access, post-accident sampling, and leakage control outside containment appear in 10 CFR 50.34(f)(2)(vii), (viii), and (xxviii), respectively.

As discussed in SECY-16-0012, "Accident Source Terms and Siting for Small Modular Reactors and Non-Light Water Reactors," dated February 7, 2016, the Commission has been considering the use of design-specific mechanistic accident source terms for SMRs. The Commission has stated that SMR applicants can use modern analysis tools to demonstrate quantitatively the safety features of those designs. Proposed design-specific accident source terms for light-water SMRs may not necessarily follow all the specific guidance that currently pertains to large LWRs.

3.1 Relevant Guidance

- NUREG-0800 (Reference 9) supplies review guidance that the staff finds acceptable in meeting the applicable regulatory requirements. The NUREG-0800 sections that contain guidance relevant to this review are Section 2.3.4, "Short-Term Atmospheric Dispersion Estimates for Accident Releases"; Section 6.5.2, "Containment Spray as a Fission Product Cleanup System"; Section 6.5.3, "Fission Product Control Systems and Structures"; and Section 15.0.3, "Design Basis Accident Radiological Consequences of Analyses for Advanced Light Water Reactors."

- NuScale DSRS, Section 15.0.3, "Design Basis Accident Radiological Consequence Analyses for NuScale SMR Design," issued June 2016 (ADAMS Accession No. ML15355A341).
- RG 1.183, "Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors," issued July 2000, provides guidance on acceptable assumptions and methodology for evaluating the radiological consequences of DBAs for LWRs, including the development of accident source terms and radiation sources for use in environmental qualification assessments for structures, systems, and components.
- RG 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Revision 1, issued November 1982, provides guidance on appropriate dispersion models for estimating offsite relative concentrations (χ/Q values) as a function of downwind direction and distance (i.e., at the EAB and LPZ) for various short-term periods (up to 30 days) after an accident.
- RG 1.194, "Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants," issued June 2003, discusses acceptable approaches for estimating short-term (i.e., 2 hours to 30 days postaccident) average χ/Q values in the vicinity of buildings at control room ventilation air intakes and at other locations of significant air inleakage to the control room envelope resulting from postulated accidental radiological airborne releases.
- RG 1.23, "Meteorological Monitoring Programs for Nuclear Power Plants," Revision 1, issued March 2007, includes guidance on the measurement and processing of onsite meteorological data for use as input to atmospheric dispersion models in support of plant licensing and operation.
- NUREG/CR-6331, "Atmospheric Relative Concentrations in Building Wakes," Revision 1, issued May 1997 (Reference 7), is the user's manual for the NRC-sponsored ARCON96 dispersion model, which is referenced in RG 1.194.
- NUREG/CR-2858, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," issued November 1982 (Reference 8), is the user's manual for the NRC-sponsored PAVAN dispersion model, which implements the guidance in RG 1.145.

4.0 Technical Evaluation

The following section of this SER documents the staff's evaluation of NuScale's proposed accident source term methodology, concentrating on the unique aspects of the methodology. The staff provides its findings on the 15 specific positions for which NuScale requested staff approval, as listed in Section 2.0 of this SER. The staff's evaluation of Position 2 on use of the ARCON96 methodology is discussed in Section 4.2 of this SER. The staff's evaluation of Positions 1, 3-10, and 14, which apply to the CDE, is discussed in Section 4.1.1 of this SER. The staff's evaluation of Positions 11-13 and 15 is discussed in Section 4.1.2 of this SER.

4.1 Accident Radiological Consequence Analyses

Section 3.0 of the topical report provides an overview of the methodology used to develop radiological source terms and perform calculations for the accident radiological consequence analyses for the NuScale SMR design. As compared to accident source term and analysis methods used for licensing of other LWR designs, NuScale's topical report describes unique methodologies in the following areas:

- atmospheric dispersion;
- iodine spike DBST;
- CDST;
- containment aerosol generation and removal; and
- post-accident pH_T

The accidents evaluated for radiological consequences at the EAB, LPZ, in the control room, and the TSC are based on the traditional DBAs evaluated for pressurized-water reactors (PWRs), as described in SRP Chapter 15 and RG 1.183. For the NuScale SMR design, these accidents are the MSLB outside containment, REA, FHA, SGTF, and the failure of small lines carrying primary coolant outside containment. For these accidents, the analysis methods conform with guidance in RG 1.183. Because the NuScale SMR design is an integral PWR with light water as the moderator and coolant and uses a fuel design similar to that of large PWRs, the staff finds the selection of accidents and the use of methods and assumptions consistent with the guidance in RG 1.183 to be acceptable.

NuScale has proposed unique analysis methods for two additional accident dose assessments. In lieu of using the RG 1.183, Appendix A, assumptions to evaluate a LOCA to show compliance with the regulatory criteria, TR-0915-17565 describes the methods for evaluating a CDE and the related CDST. In addition, the topical report describes the methods to develop a NuScale-specific iodine spike DBST to evaluate the radiological consequences of a surrogate bounding DBST to use in the evaluation of environmental qualification of equipment, as well as in the evaluation of doses at the EAB, LPZ, control room, and TSC.

4.1.1 Core Damage Event

NuScale postulates a CDE to show compliance with the regulatory requirements listed in Section 2.0 of this SER. The CDE is not a single specific accident scenario. The CDST associated with the CDE is composed of key radiological release and transport parameters, derived from a range of accident scenarios that result in significant damage to the reactor core with subsequent release of appreciable quantities of fission products into the containment. The CDST is used as input to radiological consequence assessments.

In Section 2.2 of the topical report, NuScale describes its basis for treating the CDE as a beyond-design-basis event for the NuScale SMR design. In this case, the CDST used to evaluate the radiological consequences of the CDE is based on severe accident scenarios that are related to beyond-design-basis events. The topical report proposes that the CDE radiological consequence evaluation will be explicitly defined as "beyond design basis" for the NuScale design.

NuScale categorizes a core melt accident as a beyond-design-basis event in its submissions, based on the attributes of the NuScale design in comparison to the relevant dose evaluation requirements and related NRC policy and guidance. The applicable NRC regulations do not require classification of source terms as "design basis" or "beyond design basis" to demonstrate compliance with the requirements. Therefore, the staff has determined that the classification of a core melt accident as a beyond-design-basis event for the NuScale design is not material to the staff's findings under these regulations. Therefore, the staff does not make a finding on Position 1 in Section 1.2 of the topical report regarding treatment of the CDE as a beyond-design-basis event for the NuScale design. For additional information on this topic, see staff's discussion of these terms in SECY-19-0079 (ADAMS Accession No. ML19107A455).

4.1.1.1 Accident Scenario Selection for Core Damage Source Term

RG 1.183, Regulatory Position 2, describes the attributes of an acceptable accident source term that is different from the source term specified in RG 1.183. To be considered acceptable for use in siting and safety analyses for licensing applications for power reactors, the accident source term should be based on major accidents, hypothesized for the purposes of design analyses or consideration of possible accidental events that could result in hazards not exceeded by those from other accidents considered credible. The source term must address events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products. In addition, the accident source term is not based on a single accident scenario but instead represents a spectrum of credible severe accident events. Risk insights may be used, not to select a single risk-significant accident, but rather to establish the range of events to be considered. Relevant insights from applicable severe accident research on the phenomenology of fission product release and transport behavior may be considered.

A key aspect of defining an accident source term is that the severity of the accident or group of accidents to be considered must be decided. To develop a revised accident source term for LWRs, NUREG-1465 (Reference 2) considered a range of accidents for several operating reactors, including severe accidents. NUREG-1465 defined the release in terms of four release phases: gap, early in-vessel, ex-vessel, and late in-vessel. NUREG-1465 developed values for its release characteristics based on results from NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," issued December 1990 (Reference 5), as well as the Source Term Code Package and MELCOR simulations. For use in licensing analyses, RG 1.183 then selected the first two phases for DBA dose analysis as being consistent with the regulatory requirements for siting and safety analyses. Given that the NuScale SMR is an LWR design, the staff used the information in NUREG-1465 pertaining to development of LWR accident source terms as a reference in its review of the NuScale accident source term methodology.

Sections 4.2 through 4.5 of the topical report describe the methodology for developing the CDST to be used to evaluate the CDE. The topical report states that the CDST is based on a major accident, postulated for the design analysis, and that the methodology to develop the CDST addresses events that involve a substantial meltdown of the core with the subsequent release of appreciable quantities of fission products. This is consistent with the requirements described in Footnote 3 to 10 CFR 52.47(a)(2)(iv) for the fission product release used in assessment of accident radiological consequence and is therefore acceptable to the staff.

As described in Section 4.2 of the topical report, the CDST is derived from a spectrum of surrogate accident scenarios that are indicative of the accidents associated with the regulatory description of the fission product release. The NuScale methodology considers a range of accidents for its design in the development of the CDST. The accident scenario selection is to be informed by the NuScale PRA and is intended to be representative or bounding of a dominant majority of intact containment CDEs for the NuScale nuclear power module. The methodology states that relative risk insights are used to establish a range of events to be considered for the CDE radiological consequence analysis. A subset of Level 1 PRA sequences are used to select a spectrum of surrogate accident scenarios to be used in the development of the CDST. These surrogate accident scenarios are single module internal events at full power that result in significant core damage and assume an intact containment. This consideration of a range of accidents is consistent with the LWR accident source term in NUREG-1465, as well as the guidance in RG 1.183 on accident scenarios used in the development of acceptable accident source terms. Therefore, the accident scenario selection methodology and the consideration of a range of accident scenarios is acceptable to the staff.

4.1.1.2 Core Damage Source Term Radiological Release to Containment

Section 4.2.1 of the methodology topical report states that NuScale-specific accident analyses using the severe accident code MELCOR (Reference 13) are performed to calculate the timing and magnitude of the radiological release from the failed fuel to the containment for the selected core damage surrogate accident scenarios. For each scenario, the onset time for release of fission products from the fuel gap and the duration of the gap and early in-vessel releases are calculated. Release of radionuclides to the containment is expressed as fractions of total core inventory of that radionuclide or grouping of chemically similar radionuclides, or “release fractions.” The minimum release onset time, associated release duration (i.e., for the same scenario), and the median release fractions determined from the spectrum of surrogate accident scenarios are used in the CDST.

For comparison, NUREG-1465 states the following:

> The release fractions for the source terms presented in this report were intended to be representative or typical, rather than conservative or bounding values, of those associated with a low-pressure core melt accident, except for the initial appearance of fission products from failed fuel. The release fractions are not intended to envelope all potential severe accident sequences, nor to represent any single sequence.

NUREG-1465 release fractions are mean values over all accidents from NUREG-1150, which were adjusted to reflect public comments and additional MELCOR calculations available after the issuance of draft NUREG-1465. Adjustments included reducing tellurium, barium, and strontium release fractions and changing nonvolatile radionuclide release fractions to the 75th-percentile value instead of the mean.

Because the use of median release fractions is not consistent with NUREG -1465, and because the staff had related questions on the effect of uncertainty in the fraction of the core that is predicted to overheat and release fission products, the staff evaluated the methodology in an integrated fashion for the CDE analysis, as described below in Section 4.1.1.4 of this SER. Based on the staff’s independent confirmatory analyses, as described in Section 4.1.1.4 of this SER, the staff finds it acceptable to use median release fractions from a spectrum of surrogate accident scenarios to develop the CDST. Therefore, the staff finds Position 5 in Section 1.2 of TR-0915-17565 acceptable.

4.1.1.2.1 Core Damage Source Term Release Onset and Duration

NuScale’s methodology selects the earliest time of appearance of fission products within containment as the earliest time calculated for fuel cladding rupture in its MELCOR severe accident simulations for the range of scenarios. NUREG-1465 selected the earliest time of appearance of fission products within containment based on the earliest time calculated for failure of a fuel rod, given a LOCA that results in core damage. In contrast with NuScale's methodology, NUREG-1465 is based on conservative assumptions such as the fuel rod being operated at the maximum peaking factor permitted by the plant Technical Specifications (TS). Although NuScale's methodology for selecting the gap release start time could result in longer gap release start times than NUREG-1465 methodology, it is unlikely to significantly affect the dose assessment for two reasons. The EAB dose assessment uses the worst two hours of the accident, consistent with the guidance in RG 1.183 in order to meet the regulatory requirement that the dose at the EAB is evaluated for any two-hour period. The topical report approach, together with the applicant's use of the worst two hours for the EAB radiological consequence assessment, is consistent with previous implementation of NUREG-1465 in Regulatory Guide 1.183. Therefore, the staff finds the applicant's approach acceptable with respect to the release onset timing.

NuScale's methodology selects the release duration as the shortest of the release durations from any of the scenarios. The staff finds that that choice of the shortest release duration is consistent with the discussion in NUREG-1465 and is conservative with respect to the effects of radioactive decay and mitigation. In addition, the staff evaluated the methodology's choice of release duration in an integrated fashion for the CDE analysis, as described below in Section 4.1.1.4 of this SER. Based on the staff's independent confirmatory analyses, as described in Section 4.1.1.4 of this SER, the staff finds the applicant's approach acceptable with respect to the release duration. Based on the staff's finding that NuScale's methodology to determine the CDST release onset and duration is acceptable, the staff finds Position 4 in Section 1.2 of TR-0915-17565 acceptable.

Section 4.2.2, "Core Damage," of the topical report describes the basis for [[

]] Therefore, the staff finds Position 3 in Section 1.2 of TR-0915-17565 to be acceptable.

4.1.1.2.2 Core Damage Source Term Radionuclide Groups and Iodine Chemical Form

NuScale's methodology uses radionuclide groupings from SAND2011-0128, which are different from those listed in RG 1.183, Table 5. SAND2011-0128 is a report prepared to aid the NRC staff in developing accident source terms for LWRs. As stated in the topical report, this radionuclide grouping represents the current approach used in severe accident progression analyses. No chemical elements are added or removed as compared to those listed in RG 1.183; instead some chemical elements are reassigned to different groups. The staff agrees that the radionuclide groupings from SAND2011-0128 are consistent with the state-of-the-art in severe accident modeling. Therefore, the staff finds Position 6 in Section 1.2 of TR-0915-17565 to be acceptable.

NuScale's methodology assumes the same iodine chemical form fractions as NUREG-1465 and RG 1.183. PHEBUS tests performed subsequent to the issuance of NUREG-1465 demonstrated that the NUREG-1465 iodine chemical form fractions are conservative (see ADAMS Accession No. ML003744641). Also, design differences between the NuScale SMR and large LWRs are not likely to produce different iodine chemical form fractions. Therefore, the staff finds the applicant's approach regarding iodine chemical form fractions to be acceptable.

4.1.1.3 Fission Product Transport and Removal

The NuScale accident source term methodology includes modeling assumptions for fission product transport and removal within the containment. Phenomena such as iodine re-evolution from water inside containment and in-containment natural aerosol removal processes are considered. Staff's evaluation of these modeling assumptions is below.

4.1.1.3.1 Post-Accident pH_T Calculation

The topical report, Section 4.4, "Post-Accident pH_T," describes the methodology used to evaluate the post-accident temperature-dependent pH (pH_T) in coolant water inside the containment following an event resulting in significant core damage, such as the CDE. NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants," states that the iodine released from the damaged core to the containment after a LOCA is composed of 95 percent cesium iodide, which is a highly ionized salt, soluble in water. Iodine in this form does not present any radiological concerns since it remains dissolved in the water and does not enter the containment atmosphere. However, in the radiation field in the containment, some of this iodine could be transformed from the ionic to the elemental form, which is scarcely soluble in water and can therefore be released to the containment atmosphere. Conversion of iodine to the elemental form depends on several parameters, of which pH is very important. Maintaining the pH basic in the water inside containment will ensure that this conversion will be minimized.

The staff reviewed Section 4.4 of the topical report using SRP Section 6.5.2, "Containment Spray as a Fission Product Cleanup System," and the information in NUREG/CR-5950, "Iodine Evolution and pH Control," issued December 1992 (Reference 6), as general background on the underlying basis for a method that the staff would find acceptable. Section 4.4 includes a summary of acids and bases and their sources in the NuScale design that are expected to enter the coolant and influence the pH_T during a postulated significant core damage accident. The method used in Section 4.4 is consistent with the information in SRP Section 6.5.2 and NUREG/CR-5950. The methodologies provided are also consistent with the guidance on evaluation of coolant chemistry for the purposes of modeling fission product transport and removal in DBA dose analyses given in RG 1.183. Therefore, the staff finds the proposed methodology to determine pH_T described in Section 4.4 of the topical report to be acceptable.

4.1.1.3.2 Natural Aerosol Removal Processes in Containment

Section 4.2 of the topical report describes the modeling of aerosol removal in containment through natural deposition phenomena using the aerosol removal code STARNAUA. STARNAUA includes models for aerosol removal phenomena including sedimentation, diffusiophoresis, thermophoresis and hygroscopicity [[
]] While the staff has reviewed previous licensing applications for new reactors that used STARNAUA, the staff has not reviewed STARNAUA itself for acceptability.

Similarly, the staff did not review the STARNAUA code for acceptability as part of the review of this topical report. As described in Position 3.2 of Appendix A to RG 1.183, reduction of airborne radioactivity in the containment by natural deposition may be credited. In previous LWR design certification application reviews, the staff found credit for sedimentation, diffusiophoresis and thermophoresis acceptable, including in one case the output from STARNAUA analyses by that applicant (see, for example, Section 15.3.6 of NUREG-1793). The staff found that the topical report methodology describes the modeling of applicable aerosol natural deposition phenomena in containment. The staff also evaluated the methodology's assumptions in the modeling of aerosol natural deposition in an integrated fashion for the CDE analysis, as described below in Section 4.1.1.4 of this SER. Based on the staff's independent confirmatory analyses, as described in Section 4.1.1.4 of this SER, and consistent with staff's acceptance in previous LWR design certifications, the staff finds Position 7 in Section 1.2 of TR-0915-17565 acceptable.

Applying credit for aerosol removal through natural processes requires input from thermal hydraulic and aerosol behavior models. The basis document defining the revised accident source term, NUREG-1465, does not specify an associated thermal hydraulic scenario, or methodology or acceptance criteria for aerosol removal. The alternative source term regulatory guidance, RG 1.183, also does not specify these items. NUREG-1465 describes a source term that was derived from an examination of a set of severe accident sequences for LWRs and is intended to be representative or typical and does not imply a specific scenario, much less the worst case. NuScale's methodology calculates aerosol removal coefficients in the STARNAUA code, with the thermal hydraulic data associated with the surrogate accident scenario with the minimum time to core damage used as input. The staff evaluated the methodology's choice of thermal hydraulic data in an integrated fashion for the CDE analysis, as described below in Section 4.1.1.4 of this SER. Based on the staff's independent confirmatory analyses, as described in Section 4.1.1.4 of this SER, the staff finds the methodology's approach with respect to thermal hydraulic conditions for modeling of aerosol natural deposition to be acceptable. Therefore, the staff finds Position 8 in Section 1.2 of TR-0915-17565 acceptable.

NuScale's methodology does not place an upper limit of the iodine decontamination factor for aerosol removal through natural processes. Instead, NuScale limits iodine removal by assuming 5% of the iodine is vapor and remains airborne and available to leak from the containment for the entire accident duration of 30 days. PHEBUS tests showed long-term persistent iodine airborne concentration of 0.1%. This is consistent with the guidance in RG 1.183 on estimation of fission product removal by calculation of time-dependent airborne aerosol mass. In addition, NuScale's methodology conservatively does not take credit for elemental iodine (vapor) removal. Therefore, the staff finds the methodology's approach acceptable. In addition, the staff finds Position 9 in Section 1.2 of TR-0915-17565 acceptable.

Section 4.3.6 of the topical report provides [[

]] The staff evaluated the potential effect of revaporization within the containment on the CDST in an integrated fashion for the CDE analysis, as described below in Section 4.1.1.4 of this SER. Based on the staff's independent confirmatory analyses, as described in Section 4.1.1.4 of this SER, the staff finds the methodology's approach with respect to aerosol resuspension and revaporization within the containment to be acceptable. Therefore, the staff finds Position 10 in Section 1.2 of TR-0915-17565 acceptable.

Section 4.4.6 of the topical report provides NuScale's basis for assuming that iodine re-evolution does not need to be explicitly included in the CDE dose analysis calculation for pH_T values of 6.0 or greater. NuScale estimated the amount of iodine re-evolution using Figure 3-1 of NUREG/CR-5950 to show that less than 1 percent of the aqueous iodine is converted to elemental iodine for a pH_T value of 6.0. The methodology considers this amount to be negligible, considering the overall modeling of iodine in containment. RG 1.183 provides that iodine re-evolution need not be considered from in-containment water pools with a pH of 7 or greater, based on the information in NUREG/CR-5950. Based on the topical report discussion of NUREG/CR-5950, the staff evaluated the methodology's approach to iodine re-evolution and finds it consistent with guidance and therefore, acceptable. Therefore, the staff finds Position 14 in Section 1.2 of TR-0915-17565 to be acceptable.

4.1.1.4 Independent Confirmatory Analysis for Positions 3, 4, 5, 7, 8, and 10

Feedback among the physical phenomena in the following positions prevents the staff from evaluating technical adequacy of each position individually. Therefore, the staff evaluated these positions as a group through integrated confirmatory analysis using MELCOR.

Position 3: [[ ]]

Position 4: Release timing from scenario with earliest release

Position 5: Median release fractions

Position 7: Using STARNAUA to predict aerosol deposition

Position 8: Using thermal hydraulic conditions as input to STARNAUA from scenario with earliest release

Position 10: Aerosol resuspension and revaporization in containment

The containment leaks at its design basis leak rate for the first 24 hours after the start of core damage and half the design basis leak rate after that. Although this position was not given a number in the topical report, the staff evaluated its acceptability in an integrated manner with Positions 3, 4, 5 ,7, 8 and 10.

5% of the iodine is assumed to be gaseous and not deposit in containment. Although this position was not given a number in the topical report, the staff evaluated its acceptability in an integrated manner with Positions 3, 4, 5 ,7, 8 and 10.

The staff began by reviewing the applicant's methodology for scenario selection. The staff evaluated whether the applicant's MELCOR simulations covered the credible core-damage sequences. The conditions needed to lead to core damage include a sustained loss of cooling. Such conditions could occur in the NuScale SMR design as a result of a leakage from the reactor coolant system and emergency core cooling system (ECCS) failure. One type of core-damage accident scenario includes a break at a higher elevation in the reactor pressure vessel (RPV) such as a failed-open reactor vent valve (RVV). In this case, coolant cannot return to the RPV because the break location is at the top of the RPV. Another type of core-damage accident scenario includes a break at a lower elevation in the RPV such as a failed-open reactor RRV. Coolant can reenter the RPV in this case because the break elevation is below the water level in containment produced by discharge of the RPV inventory into the containment. In the topical report's example analysis, the applicant selected five scenarios, of which three had a break at a higher elevation in the RPV, and two had a break at a lower elevation in the RPV. The five scenarios cover most of the core damage frequency (CDF) for the NuScale SMR design.

To evaluate the CDST methodology, the staff performed independent confirmatory analysis using MELCOR and the dose analysis code RADTRAD (Reference 14). The staff independently developed a MELCOR input model using plant design data provided by the applicant. The staff applied its model to the following two scenarios in the applicant's PRA:

- LEC-06T-00: A stuck-open RVV with subsequent opening of the remaining two RVVs. This scenario is representative of scenarios with a break at a high elevation in the RPV such that steam is discharged through the break. Liquid water cannot return to the RPV because the break location is at the top of the RPV.

- LCC-05T-01: Chemical and volume control system (CVCS) line break inside containment with subsequent opening of the three RVVs. This scenario is representative of scenarios with a break at a low elevation in the RPV such that liquid water is discharged through the break. Liquid water cannot return to the RPV because the CVCS piping rupture is in the containment upper plenum.

The staff's independent MELCOR confirmatory analysis is documented in RES/FSCB 2019-01, "Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor," April 2019 (ADAMS Accession No. ML19205A016). The staff used its MELCOR-predicted releases to the environment from these two scenarios as input to dose analyses using the RADTRAD computer code. The staff's independent RADTRAD analysis is documented in RES/FSCB 2019-03, "Independent Confirmatory Analysis for NuScale Offsite Radiological Consequence Assessment," August 2019 (ADAMS Accession No. ML19240A046, not publicly available). While the staff's independent MELCOR and RADTRAD analyses predict doses up to a factor of 2.5 higher than the applicant's example calculations, the staff finds that the difference between the staff's and applicant's results are generally within the uncertainty margin of design-basis accident calculations. Therefore, the staff finds the use of the above eight positions acceptable as a group. Considering the applicant's description of the technical bases for the core damage assumptions, aerosol transport and removal within the containment, along with the staff's analysis of the sensitivity of the overall dose results to the uncertainty in the dose analysis modeling of these phenomena, the staff finds that the methodology to develop the CDST and calculate the radiological consequences of the CDE is acceptable.

4.1.2 Design-Basis Accident Source Terms

Section 3.2 of the topical report describes the radiological consequence analysis methodology for the REA, FHA, MSLB, SGTF, failure of small lines carrying coolant outside containment, and the iodine spike DBST. In general, the NuScale methodology for each of these events is based on the guidance in RG 1.183, with adjustments as justified by the NuScale SMR design. Because the NuScale SMR design is an integral PWR with light water as the moderator and coolant and uses a fuel design similar to that of large PWRs, the staff finds the use of methods and assumptions consistent with the guidance in RG 1.183 acceptable for the evaluation of DBA source terms listed above for the NuScale SMR design. The following subsections of this SER discuss the staff's evaluation of NuScale's specific positions on the proposed use of RG 1.183 guidance for the NuScale design. The staff's evaluation of the proposed methodology for specific accident analyses includes additional considerations related to the use of the guidance in RG 1.183, as described below.

4.1.2.1 Iodine Spike Design-Basis Source Term

The iodine spike DBST is unique to the NuScale SMR design, derived from the assumption of a generic failure occurring that results in the release of all primary coolant from the RCS to the containment. The iodine spike DBST is a postulated surrogate accident source term that is intended to bound the radiological consequences of a spectrum of events that result in primary coolant being released to an intact containment. Although RG 1.183 does not explicitly describe such a source term, NuScale used guidance that assumes primary coolant as the source of radionuclides released to the environment (the release does not involve core or fuel damage), as far as applicable.

As described in Section 3.2.6 of the topical report, the iodine spike DBST assumes that the entire radionuclide activity within the primary coolant is instantaneously available within the containment free volume. No fission product removal in containment is modeled. The staff finds these assumptions to be conservative and therefore, acceptable. Consistent with guidance in RG 1.183, two iodine spiking cases model the radionuclide inventory in the primary coolant. Section 4.1.2.1.2 of this SER discusses the staff's evaluation of primary coolant iodine spiking. Containment leakage rates are based on the technical specification (TS) design containment leak rate, which is consistent with the guidance in RG 1.183 for the LOCA and therefore, acceptable to the staff. All other assumptions are the same as described in the NuScale methodology for other DBAs and are also consistent with the guidance in RG 1.183. Therefore, the staff finds the assumptions for the postulated iodine spike DBST to be acceptable.

4.1.2.1.1 In-Containment Radiation Source for Environmental Qualification

This section assumes that the changes proposed in response to RAI 9690, Questions 01.05-39 and 01.05-41, dated July 31, 2019 (ML19212A801) have been incorporated into the TR. The NRC staff will confirm the changes are incorporated in the approved version that will be issued by the applicant.

10 CFR 50.49(e)(4) requires, in part, that the radiation environment considered in the electric equipment qualification program must be based on the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional. In addition, 10 CFR Part 50, Appendix A, GDC 4 requires, in part, that structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents; including loss-of-coolant accidents. Previous LWR applications have assumed significant core damage in order to address the radiological aspects of these requirements, which is consistent with RG 1.89 and 1.183, Appendix I. However, in TR- 0915-17565, NuScale indicated that there are no credible design basis events in the NuScale design that result in substantial core damage. The staff reviewed the potential accident scenarios in the NuScale FSAR and determined that there are no design basis events in the NuScale design that result in core damage. Therefore, the staff has determined that the source term used to evaluate compliance with 10 CFR 50.49(e)(4) in the NuScale design does not need to address core damage (as described in SECY-19-0079). Similar to 10 CFR 50.49(e)(4), since loss of coolant accidents and other design basis events do not result in core damage in the NuScale design and since the design of equipment under 10 CFR Part 50, Appendix A, GDC 4 for other parameters (such as pressure and temperature) are not evaluated using source terms that consider core damage, the staff has determined that core damage need not be assumed in addressing the radiological equipment qualification aspects of

10 CFR Part 50, Appendix A, GDC 4. However, while core damage is not considered in addressing the requirements of 10 CFR 50.49(e)(4) and GDC 4, the staff notes that a core damage equipment survivability analysis is needed for equipment which is required to function to withstand core damage events, as required by 10 CFR 52.47(a)(23) and 10 CFR 50.44 and as provided in SECY-90-016, SECY 93-087, and the associated SRM for SECY 93-087 (ML003708056). Information on equipment survivability for the NuScale design is provided in Chapter 19 of the NuScale FSAR.

Since there are no design basis events that result in core damage in the NuScale design, the applicant proposes using the iodine spike design basis source term as the bounding source term for environmental qualification to meet 10 CFR 50.49(e)(4). Appendix B of the TR provides the methodology used in calculating the dose for environmental qualification inside containment and under the bioshield. While the iodine spike source term is not based on a specific accident, a rapid increase (or spike) in reactor coolant radionuclide concentrations is known to occur following transients at nuclear power plants and the spiking of iodine is discussed in RG 1.183. The remainder of this section discusses the requirements of 10 CFR 50.49(e)(4) and GDC 4 and the staff's evaluation of the iodine spike source term for environmental qualification and the dose methodology discussed in Section 3.2.6 and Appendix B of the TR.

The iodine spike source term proposed by NuScale for the most severe design basis event inside containment and under the bioshield for equipment qualification includes an iodine spike factor of 500 rate for 8 hours. The iodine spike factor is consistent with the maximum value in RG 1.183 and, therefore, the spiking factor of 500 for iodine is acceptable. However, while RG 1.183 does not provide guidance for the spiking of radionuclides other than iodine, RG 1.183, Appendix I indicates that a core damage source term should be assumed and other source terms (including an iodine spike source term, for example, as a result of a main steam line break) should only be considered for equipment where the core damage source term is not bounding. As discussed above, a core damage source term was not used for equipment qualification in the NuScale design. Therefore, other radionuclides, besides iodine, would also be expected to increase following a transient. The staff evaluated other conservative assumptions in developing the source term and equipment qualification dose rates for equipment qualification doses inside the CNV, RPV, and under the bioshield. [[

]] In reviewing all the assumptions for calculating doses inside of containment and under the bioshield in the topical report, the staff found that while some of the assumptions were conservative, there was not enough information for the staff to conclude that the conservatisms bound the potential increase in the source term that could occur due to the spiking of other radionuclides (besides iodine) following a design basis accident or transient.

The staff performed an independent calculation and estimated the dose rates inside containment and under the bioshield that would be expected using conservative assumptions to account for the potential spiking of radionuclides other than iodine. The staff found that there is sufficient margin between the methods used by the staff and the total integrated normal operational dose values found in Columns A through G of Table 3C-6 of the NuScale FSAR. Columns A through G of Table 3C-6 provide the normal operational total integrated doses for regions inside of containment and under the bioshield in the NuScale design. The normal operation total integrated doses in Table 3C-6 were found to be acceptable by the staff in the review of the FSAR for normal operation sources and dose rates. The normal operation doses in upper areas of containment (and other areas away from the core) are higher in the NuScale design than they are in large LWRs because of unique features in the NuScale design, mainly the very small containment which allows significant neutron doses throughout containment. In large LWR designs, neutron doses are normally insignificant in the outside and upper portions of containment due to the significant shielding between the core and these areas. In addition, in the NuScale design, the entirety of containment is impacted by normal operational gamma doses, including N-16. Likewise, the total integrated doses in areas under the bioshield during normal operation for the NuScale design are significant, including significant dose from neutrons due to the proximity to the sources inside containment. Based on this, the staff determined that the equipment qualification normal doses inside of containment and under the bioshield provide sufficient margin over the accident doses in the NuScale design that accident doses would not be a significant contributor to the total integrated dose. The staff concludes this is the case, even if the spiking other radionuclides were considered, due to the high total integrated doses in these areas during normal operation and because the most severe design basis accident in the NuScale design, for the purposes of meeting 10 CFR 50.49(e)(4) and GDC 4, does not include core damage. Based on this, the staff finds it acceptable to use the iodine spike source term methodology and the environmental qualification dose methodology described in Appendix B of the topical report for calculating environmental qualification doses to these areas in the NuScale design.

In addition, for some design basis events, such as a main steam line break, the affected line would be expected to be isolated shortly after the initiation of the accident, which would be expected to result in a transient increase in the source term for areas near the main steam line. As a result, the staff finds it to be acceptable to use the 8-hour iodine spike source term for main steam line break and other accidents for equipment qualification for those areas where there would be a transient source term increase following an accident.

As specified in Section 6.0 of this SER, the staff has set conditions and limitations for the use of the topical report iodine spike source term and Appendix B methodology because the staff's assessment shows that the accident doses are not a significant contributor to the total integrated dose in comparison to the normal operation doses for areas inside containment and under the bioshield, and may not apply for a reactor of a different design or a reactor with differences in size or geometry than the design docketed under Docket Number 52-048. The staff approves the use of the topical report for calculating main control room habitability and offsite radiological consequences of design basis accidents because the iodine radionuclides are expected to dominate these source terms. In addition, the core damage source term is used for these assessments, which is bounding. The staff also approves the methodology for evaluating the environmental doses outside containment for only design basis accidents that would result in a transient spiking source term outside of the NPM area (such as main steam line break accidents). The staff does not approve this source term for a situation in which the fluid is intentionally brought outside containment or for evaluating the dose to individuals located in the vicinity to the radioactive material in lines outside of containment. Staff reached this conclusion because there could be significant dose contribution to these areas from radionuclides that could spike (or increase) following a transient or accident. However, the staff notes that NuScale is exempt from post-accident sampling, post-accident sampling is not identified in the NuScale FSAR, and there are no situations where fluid is intentionally removed from containment in the NuScale design during design basis accidents. As a result, the staff did not assess conditions in which design basis accident post-accident fluid is intentionally removed from containment. Hydrogen and oxygen monitoring is a design feature of the NuScale design and is required by 10 CFR 50.44, "Combustible Gas control for nuclear power reactors." Systems associated with this activity are located outside of the NPM area. The dose from this activity is evaluated using a core damage source term and is discussed in Chapters 6, 9, 12, and 19 of the NuScale FSAR.

With the conditions and limitations specified in Section 6.0 of this SER, the staff finds the iodine spike design basis source term and dose methodology provided in TR-0915-17565 acceptable for the reasons discussed above.

4.1.2.1.2 Primary Coolant Iodine Spiking

The MSLB, SGTF, and failure of small lines carrying primary coolant outside containment (small line break) do not result in fuel damage, so the source of radioactivity for potential release to the environment is the radioactivity in the primary coolant. This radioactivity is released to the secondary coolant through primary-to-secondary steam generator tube leakage (or steam generator break flow for the SGTF), and eventual release to the environment, either through the main break or valve leakage to the reactor building. The topical report methodology proposes to conform to RG 1.183 guidance on the assumptions for primary coolant initial activity concentration based on TS limits, which is acceptable to the staff.

The topical report methodology also proposes to conform to the RG 1.183 assumptions on modeling of iodine spiking in the primary coolant. For the pre-incident iodine spike, the methodology assumes that the primary coolant iodine concentration is elevated, consistent with the allowable level of primary coolant specific activity in the TS. The staff finds this acceptable for the evaluation of dose to an individual offsite, in the control room, or in the TSC, because the analysis input is related to TS that control the level of radioactivity in the coolant, consistent with guidance in RG 1.183.

The topical report methodology proposes to use the iodine appearance rate spiking factors for coincident iodine spiking from RG 1.183 for the NuScale design. The RG 1.183 coincident iodine spiking assumptions are nonmechanistic values that bound an expected temporary increase in the primary coolant iodine concentration, based on PWR coolant measurements. The coincident iodine spiking case assumes that the primary coolant iodine concentration is at the TS equilibrium level at the initiation of the accident, and the iodine concentration increases and then returns to the initial level over a defined duration (typically 8 hours for large PWRs). This iodine spike results from increased leakage from intact fuel after a sudden and large decrease in power and RCS pressure. Because the NuScale design uses fuel that is similar to PWR fuel and uses light water in a pressurized primary coolant system, the staff finds that the conditions in the NuScale primary coolant are similar to those used to develop the assumptions in RG 1.183. Therefore, the assumptions about coincident iodine spiking are acceptable for the evaluation of dose to an individual offsite, in the control room, or in the TSC. Based on the discussion above, the staff finds Position 11 in Section 1.2 of TR-0915-17565 acceptable, when limited to the evaluation of dose to an individual off site, in the control room, or in the TSC.

4.1.2.2 Secondary Coolant Modeling

Section 3.3.2.1 of the topical report describes that for the NuScale SMR design, the ratio of secondary coolant to primary coolant is small (approximately 1 percent). Therefore, the topical report methodology does not model the secondary coolant, including any radioactivity that may be in the secondary coolant. The staff finds this acceptable because the initial secondary coolant activity concentration is an order of magnitude less than the primary coolant activity concentration and thus would not add significantly to the radiological release from the primary coolant for any of the accidents analyzed. Based on the discussion above, the staff finds Position 13 in Section 1.2 of TR-0915-17565 acceptable.

4.1.2.3 Reactor Pool Decontamination Factor for Fuel Handling Accident

The topical report methodology uses the RG 1.183 assumption on the pool iodine decontamination factor for the FHA. RG 1.183, Appendix B, Position 2, states that if the depth of water above the damaged fuel is 23 feet or greater, the decontamination factors for the elemental and organic species are 500 and 1, respectively, giving an overall effective decontamination factor of 200 (i.e., the water retains 99.5 percent of the total iodine released from the damaged rods). NuScale design information describes the minimum reactor pool depth as greater than 23 feet above potentially damaged fuel locations as a result of an FHA. Therefore, the staff finds the use of the RG 1.183 assumption on pool iodine decontamination factor for the FHA acceptable. Based on the above discussion, the staff finds Position 12 in Section 1.2 of TR-0915-17565 acceptable.

4.1.2.4 Containment Shine

The dose from gamma radiation shine through the containment vessel, reactor pool, and through the reactor building walls and ceiling to the environment is assumed to be negligible for the NuScale design. More than half of the containment vessel is submerged in the reactor pool for the duration of the limiting accident (the CDE). The reactor pool water provides shielding, and this, along with the plant layout and additional shielding from the reactor building structures, would greatly reduce the amount of radiation to the environment outside the site, including at the EAB. In its March 22, 2017, response (ADAMS Accession No. ML17081A561) to RAI 8706, Question 01.05-23, NuScale provided information on the sensitivity of the offsite dose results to contributions from containment shine. The staff audited (ADAMS Accession No. ML17223A659) the proprietary calculations and confirmed that example calculations show that the contribution from containment shine to offsite doses is negligible. Therefore, the staff finds that the NuScale topical report methodology is acceptable with respect to the evaluation of potential containment shine dose at offsite locations. Based on the discussion above, the staff finds Position 15 in Section 1.2 of TR-0915-17565 acceptable.

4.2 Atmospheric Dispersion Factors

NuScale uses the ARCON96 computer code methodology (Reference 7) for calculating offsite atmospheric dispersion values (Position 2 in Section 1.2, “Scope,” of the topical report) rather than the computer code PAVAN (Reference 8). Both PAVAN and ARCON96 are NRC codes approved for calculating relative concentrations (also known as atmospheric dispersion factors or χ/Q values). PAVAN implements the guidance in RG 1.145, “Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,” Revision 1, for determining offsite χ/Q values at the EAB and outer boundary of the LPZ, whereas ARCON96 implements the guidance in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants,” for determining onsite χ/Q values for the control room.

Large LWR nuclear power plants typically have EAB and LPZ distances that range from 800 to 6,000 meters, whereas NuScale postulated in the DCA an EAB and LPZ at the site boundary, which is estimated to be in the range of 80 to 400 meters. The applicant contends that PAVAN is conservative, especially for the shorter EAB and LPZ distances expected to accompany COL applications that reference the NuScale SMR design. The applicant states that the ARCON96 computer code, which was developed to model shorter distances in the near vicinity of buildings typical of control room habitability dose evaluations, is more appropriate for modeling EAB and LPZ χ/Q values for the NuScale reactor.

4.2.1 Summary of Technical Information Related to Atmospheric Dispersion Modeling

Section 4.1 of TR-0915-17565 describes NuScale’s proposed methods for calculating relative concentrations for a reduced EAB and LPZ provided in a NuScale DCA or in a COL application that references a certified NuScale SMR design.

NuScale proposes using the atmospheric dispersion algorithms in the computer code ARCON96 in lieu of the atmospheric dispersion algorithms in the computer code PAVAN to calculate accident χ/Q values for radiological releases to the EAB and LPZ.

4.2.1.1 PAVAN

The PAVAN code estimates χ/Q values for various time-averaged periods ranging from 2 hours to 30 days. PAVAN’s algorithms address reduction in ground-level concentration estimates resulting from the combined effects of building wake and plume meander during the occurrence of stable atmospheric conditions and light windspeeds. PAVAN’s methodology is based on field studies conducted at two facilities during the 1970s (Reference 10). The meteorological input to PAVAN consists of a joint frequency distribution of hourly values of windspeed and wind direction by atmospheric stability class. The χ/Q values calculated by PAVAN are based on the theoretical assumption that material released into the atmosphere will be normally distributed (Gaussian) about the plume centerline. A straight-line trajectory is assumed between the point of release and all distances for which χ/Q values are calculated.

For each of the 16 downwind direction sectors (N, NNE, NE, ENE, etc.), PAVAN calculates χ/Q values for each combination of windspeed and atmospheric stability at the appropriate downwind distance (i.e., the EAB and LPZ). The χ/Q values calculated for each sector are then placed in order from the greatest to the smallest, and their associated frequencies are summed to generate a cumulative probability distribution, which is derived from joint frequency occurrences of windspeed and stabilities for each sector. PAVAN determines for each sector an upper envelope curve based on the derived data (plotted as χ/Q versus probability of being exceeded), so that no plotted point is above the curve. From this upper envelope, the χ/Q value, which is equal to or exceeded 0.5 percent of the total time, is obtained. The maximum 0.5-percent χ/Q value from the 16 sectors becomes the 0–2 hour “maximum sector χ/Q value.”

Using the same approach, PAVAN also combines all χ/Q values independent of wind direction into a cumulative probability distribution for the entire site. An upper envelope curve is determined, and the program selects the χ/Q value that is equal to or exceeded no more than

5 percent of the total time. This value is known as the 0–2 hour "5-percent overall site χ/Q value."

The user selects the larger of the two χ/Q values, either the 0.5-percent maximum sector value or the 5-percent overall site value, from the PAVAN output to represent the χ/Q value for the 0–2-hour time interval, as stated in RG 1.145. Note that this resulting χ/Q value is based on 1-hour averaged data, but it is conservatively assumed to apply for 2 hours.

To determine LPZ χ/Q values for longer periods (e.g., 0–8 hours, 8–24 hours, 1–4 days, and 4–30 days), PAVAN performs a logarithmic interpolation between the 0–2-hour χ/Q values and the annual average (8,760 hours) χ/Q values for each of the 16 sectors and the overall site. For each time period, the highest among the 16 sector and overall site χ/Q values is identified and becomes the χ/Q value for that period.

4.2.1.2 ARCON96

In the mid-1980s, the NRC staff determined that its DBA atmospheric dispersion modeling guidance, which included RG 1.145 and PAVAN, significantly overpredicted concentrations during light winds in the vicinity of buildings and embarked on a series of studies that ultimately resulted in the ARCON96 model. ARCON96 is based on field measurements taken at seven reactor sites. The downwind distances of the field measurements ranged from locations on and adjacent to buildings out to distances of 1,200 meters. The results were a set of revised diffusion coefficients that had low windspeed and building wake corrections. The resulting dispersion algorithms improved model performance by reducing overpredictions without significantly increasing underpredictions.

The staff subsequently endorsed ARCON96 in RG 1.194 as a method for determining atmospheric relative concentrations in support of design-basis radiological habitability assessments for the control room.

ARCON96 calculates hourly χ/Q values using hourly meteorological data. The hourly χ/Q values are then combined to estimate relative concentration averages for periods ranging from 2 hours to 30 days. The code implements a plume centerline Gaussian diffusion model for averaging times of 8 hours or less and implements a sector average Gaussian diffusion model for longer averaging times. Because wind direction is considered as the averages are formed, the averages account for persistence in both diffusion conditions and wind direction. Cumulative frequency distributions are prepared from the average relative concentrations, and relative concentrations that are exceeded no more than 5 percent of the time (95^{th}-percentile concentrations) are determined for the cumulative frequency distributions for each averaging period.

4.2.1.3 NARCON

[[ ]] NuScale developed the NARCON atmospheric dispersion model. NARCON is the NuScale version of ARCON96 [[ ]]

4.2.1.4 Differences between PAVAN and ARCON96 Methodologies

NuScale discusses key differences between the PAVAN and ARCON96 methodologies in Section 4.1.3 of the topical report. Key differences include the following:

- PAVAN's meteorological input is a joint frequency distribution of hourly windspeed, wind direction, and atmospheric stability data, whereas ARCON96's meteorological input is a database of hourly data.

- PAVAN and ARCON96 have different sets of atmospheric dispersion algorithms intended to address reduction in ground-level concentration estimates caused by the combined effects of building wake and plume meander during the occurrence of stable atmospheric conditions and light windspeeds.

PAVAN calculates relative concentrations that are exceeded no more than 0.5 percent of the time (99.5th-percentile concentrations) for each downwind sector and a relative concentration that is exceeded no more than 5 percent of the time (95th-percentile concentration) for all sectors combined in one run, whereas ARCON96 calculates a 95th-percentile relative concentration only for one downwind sector in one run.

4.2.1.5 Differences between PAVAN and ARCON96 Results

Sections 4.1.4 and 4.1.5 of the topical report provide comparisons (1) between calculated relative concentrations and observed concentrations as presented in the basis document for ARCON96 (Reference 11) and (2) between PAVAN and ARCON96 results.

The observed concentrations were recorded from various experiments with distances ranging from 8 to 1,200 meters, atmospheric stability classes ranging from extremely unstable to extremely stable, and windspeeds ranging from less than 1 meter per second to greater than 10 meters per second. NuScale's model comparison emphasizes low windspeed and stable conditions because concentrations predicted by PAVAN for these conditions typically provide the limiting case in evaluation of consequences of accidental releases in the vicinity of buildings.

[[

]]

4.2.1.6 Description of NuScale's ARCON96 Methodology

Sections 4.1.6 and 6.1.1 of the topical report states that a DCA or a COL application that uses the ARCON96-based methodology of this topical report (as implemented in NARCON) must satisfy the following criteria:

- [[ ]]
- [[ ]]
- [[ ]]
- A ground-level [[ ]] should be assumed.
    - [[ ]]
    - [[ ]]
    - [[ ]]
- [[ ]]
- [[ ]]

Section 4.1.3 of the topical report provides the following additional guidance for implementing the ARCON96-based NARCON methodology.

Since ARCON96 calculates a relative concentration for only one specified direction per code execution, the NuScale methodology specifies performing 16 executions of the code, one for each wind direction sector. PAVAN assumes each of the 16 direction sectors are 22.5 degrees wide, while ARCON96 allows the user to specify the width of the wind direction window in degrees. [[ ]]

NARCON is the NuScale version of ARCON96 [[ ]]

The NuScale methodology assumes that the EAB and LPZ are a uniform circle where the distance to each of the 16 direction sectors is of equal length. [[

]]

4.2.1.1 Atmospheric Dispersion Example Calculation

NuScale presents example atmospheric dispersion calculation analyses and results in Section 5.1 of the topical report to demonstrate the application of the methodology described in the topical report. These results are for illustration only; NuScale did not update these example calculation results to reflect the final version of its methodology because final design values are provided as part of the DCA. The staff used the example calculation as information in its evaluation of the proposed methodology and does not make a finding as to the acceptability of the example calculation analyses and results.

To demonstrate the application of ARCON96-based methodology, NuScale used hourly data for a 3-year span (1984 to 1986) from a National Weather Service (NWS) observation station in Sacramento, CA, in its example calculations. NuScale chose this dataset from a study of atmospheric dispersion factors for 241 sites located across the United States because the resulting atmospheric dispersion factors represented a site in the 80–90th percentile as recommended by the Electric Power Research Institute's "Advanced Light Water Reactor Utility Requirements Document," Revision 8, issued March 1999 (Reference 12). RG 1.23 classifies atmospheric stability as a function of vertical temperature difference, or delta-T. Since the NWS does not typically collect lower elevation delta-T data, NuScale used a meteorological processor program, PCRAMMET, from the U.S. Environmental Protection Agency to calculate atmospheric stability. PCRAMMET calculates atmospheric stability as a function of solar insolation and cloud cover. The program can produce different stability classes in the absence of site-specific delta-T information. The delta-T method has been known to result in a higher frequency of limiting case stable stability atmospheric conditions under which the highest ground-level concentrations occur for ground-level releases. NuScale notes that the PCRAMMET methodology was used only for illustrative purposes to select an 80-90th percentile U.S. site. This representative site is assumed to occur on flat ground with nominal surface features (i.e., default surface roughness).

As part of a 2017 audit of this topical report, the staff reviewed the NWS 1984 Sacramento meteorological dataset (ADAMS Accession No. ML17304B303). The staff found the stability class frequency distribution to be reasonable (e.g., a generally normal distribution centered on neutral (D) stability). The wind directions were somewhat bifocal, with maximums occurring with wind directions from 40 degrees (6.6 percent) and 150 degrees (5.6 percent) and a minimum with wind directions from 220 degrees (0.4 percent). The windspeed distribution (which was reported to the nearest whole knot) was typical for an NWS site using wind sensors with a high starting threshold, with 13.7 percent recorded as calm (0 knots), no recordings for 1 knot, and 0.2 percent recorded at 2 knots. The most frequent windspeed was 5 knots (13.6 percent).

Table 5-4, "Example offsite atmospheric relative concentration (χ/Q) values," of the topical report provides the resulting relative concentrations that are used in the example dose calculations in the topical report.

4.2.2 Technical Evaluation of Atmospheric Dispersion Factor Methodology

NuScale's topical report describes the applicant's methods for determining accident χ/Q values for the EAB and LPZ using a methodology that differs from the NRC's guidance. The staff reviewed the topical report in accordance with NUREG-0800, Section 2.3.4, which states that a DC applicant should include EAB and LPZ boundary χ/Q values for the appropriate time periods in the list of site parameters. This information should include the determination of conservative χ/Q values used to assess the consequences of postulated design-basis atmospheric radioactive releases to the EAB and LPZ.

a. Use of ARCON96 as an alternative methodology to PAVAN.

NuScale's justification for choosing ARCON96 instead of PAVAN to calculate offsite relative concentration values is that the EAB and LPZ boundaries for those COL applicants referencing the planned NuScale DC are expected to be smaller than those of reactors that currently operate in the United States or COL applications that have been recently approved by the NRC staff. In addition, NuScale notes that ARCON96 is already approved for use for control room calculations, as discussed in RG 1.194. In the case of a smaller EAB and LPZ boundary, on the order of 80–400 meters, NuScale plans to use its version of ARCON96, NARCON, instead of PAVAN because PAVAN would be overly conservative for the EAB and LPZ calculations.

For the reasons discussed below, the staff finds the licensee's proposal to use NARCON as an alternative to PAVAN acceptable, based on the methodology described below and with the conditions and limitations discussed in Section 6 of this SER.

b. [[

]]

The staff finds this approach acceptable because [[

]] are consistent with the guidance in RG 1.145. Note that [[

]]

NuScale developed the NARCON atmospheric dispersion model [[

]] The staff reviewed the documentation for the NARCON computer code and executed several runs as part of an audit of this topical report (ADAMS Accession No. ML17304B303). The staff found that the code can be executed [[

]]

c. [[

]]

The staff finds this criterion acceptable because [[

]]

d. [[ ]]

The staff finds this criterion acceptable because [[ ]]

e. [[ ]]

The staff finds this criterion acceptable because [[ ]]

f. [[ ]]

The staff finds this criterion acceptable because [[ ]]

g. [[ ]]

The staff finds this criterion acceptable. [[ ]]

The staff did not review in detail Sections 4.1.5 and 4.1.6 of the topical report in its decision to accept or reject the proposed methodology and did not depend on the information in these sections. ARCON96 is a general code for assessing atmospheric relative concentrations in building wakes under a wide range of situations and was approved by the NRC staff in RG 1.194 for use in performing control room atmospheric dispersion calculations. The ARCON96 dispersion algorithms are based on field measurements taken out to distances of 1,200 meters. [[ ]] Based on the discussion above, the staff finds Position 2 in Section 1.2 of TR-0915-17565 acceptable.

5.0 Staff Conclusions

The NRC staff has completed its review of the NuScale licensing TR-0915-17565, Revision 3, and concludes that, subject to the conditions and limitations specified in Section 6.0 of this SER, the methods described in the topical report are acceptable for developing accident source terms and performing accident radiological consequence analyses to be referenced by the NuScale SMR design. The staff approves Positions 2 through 15 in Section 1.2 of TR-01915-17565. The staff does not make a finding on Position 1. The staff's conclusions on specific technical topics appear in the respective technical evaluation sections of this report.

Therefore, the staff approves the use of the NuScale licensing TR-0915-17565, Revision 3, subject to the conditions and limitations specified in Section 6.0 of this SER, in support of a

NuScale SMR DC or for reference by NuScale COL holders or COL applicants, in accordance with applicable license requirements.

6.0 Conditions and Limitations

(1) The staff's approval of TR-0915-17565, Revision 3, applies only to the NuScale SMR design. The NuScale SMR design is defined as the design described on Docket Number 52-048 and subsequent revisions to that design that continue to maintain the same fundamental size, geometry, and safety features of the design docketed in 52-048. Any use in whole, or in part, for other designs would require an additional applicability review by the staff.

(2) Approved applications of the source terms described in TR-0915-17565 are limited to (1) assessments of main control room habitability and offsite radiological consequences of DBAs, and (2) the assessment of environmental qualification doses as described in Appendix B of the topical report is only for areas or components inside of the containment vessel and under the bioshield, and shine from those contained sources, and to areas outside of the NPM bay prior to the isolation of containment (assessment of equipment qualification doses from fluids intentionally removed from containment, during and following a DBA, is not an approved application).

(3) The staff makes no finding on the treatment of the CDE as a beyond-design-basis event for the NuScale design.

(4) The use of NuScale's methodology by COL applicants will require the submittal of site-specific meteorological data. The meteorological data needed by ARCON96 for χ/Q calculations include windspeed, wind direction, and a measure of atmospheric stability. These data should be obtained from an onsite meteorological measurement program based on the guidance in RG 1.23.

(5) A COL applicant referencing NuScale's design should follow the guidance in RG 1.23 for the calculation of atmospheric stability. A COL applicant should use the vertical temperature difference method to determine stability for use in relative concentration calculations. If other well-documented methodologies are used to estimate atmospheric stability (with appropriate justification), the ARCON96 model may require modification.

(6) [[
]]

(7) A COL applicant who uses this methodology is expected to evaluate the applicability of the atmospheric dispersion modeling methodology for any significant site-specific geographical features.

(8) The selection of release location affects the distance between the release point and the EAB and LPZ, which is used to calculate the offsite dispersion factor. This distance should be calculated in accordance with Regulatory Position 1.2 in RG 1.145.

7.0 References

1. NuScale, "Accident Source Term Methodology," TR-0915-17565, Revision 3, Corvallis, OR, April 21, 2019, ADAMS Accession No. ML19112A172.
2. U.S. Nuclear Regulatory Commission, "Accident Source Terms for Light-Water Nuclear Power Plants," NUREG-1465, February 1995.
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4. Sandia National Laboratories, SAND2011-0128, "Accident Source Terms for Light-Water Nuclear Power Plants Using High-Burnup or MOX Fuel," Albuquerque, NM, January 2011 (available at https://www.osti.gov/biblio/1010412).
5. U.S. Nuclear Regulatory Commission, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," NUREG-1150, December 1990.
6. U.S. Nuclear Regulatory Commission, "Iodine Evolution and pH Control," NUREG/CR-5950, December 1992.
7. U.S. Nuclear Regulatory Commission, "Atmospheric Relative Concentrations in Building Wakes," NUREG/CR-6331, Revision 1, May 1997.
8. U.S. Nuclear Regulatory Commission, "PAVAN: An Atmospheric Dispersion Program for Evaluating Design Basis Accidental Releases of Radioactive Materials from Nuclear Power Stations," NUREG/CR-2858, November 1982.
9. U.S. Nuclear Regulatory Commission, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition," NUREG-0800, June 2007.
10. U.S. Nuclear Regulatory Commission, "Technical Basis for Regulatory Guide 1.145, 'Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants,'" NUREG/CR-2260, October 1981.
11. Pacific Northwest Laboratory, "Atmospheric Dispersion Estimates in the Vicinity of Buildings," Richland, WA, January 1995.
12. Electric Power Research Institute, "Advanced Light Water Reactor Utility Requirements Document," Revision 8, Washington, DC, March 1999.
13. Sandia National Laboratories, "MELCOR Computer Code Manuals," Vol. 1: Primer and Users' Guide, Version 2.2.9541," SAND 2017-0455 O, January 2017 (ADAMS Accession No. ML17040A429)
14. US Nuclear Regulatory Commission, "SNAP/RADTRAD 4.0: Description of Models and Methods," NUREG/CR-7220, June 2016

Section B

Licensing Topical Report

Accident Source Term Methodology

February 2020

Revision 4

Docket: PROJ0769

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List of Affected Pages

Revision Number	Page Number	Explanation
3	Throughout	The terms "Category 1" and "Category 2" are no longer used. The term "design basis source term" (DBST) no longer refers to the source term formerly known as the "Category 2 source term," but rather to the source term formerly known as "Category 1 source term."
3	Throughout	Introduced the term "core damage source term" (CDST) for the source term formerly known as the "Category 2 source term."
3	Section 4.4 relocated to Section 5.8	Sensitivity examples moved to the example section.
3	Throughout	Aerosol removal methodology changes driven by eRAI No. 9224.
3	Throughout	Replaced references to "forthcoming DCD" and similar terms.
3	Throughout	Replaced "source term design basis accident" (STDBA) with "surrogate accident scenario" to emphasize the beyond-design-basis nature of the core damage event (CDE) and associated CDST.
3	Sections 1.2, 4.2.1, 4.2.3, and 4.2.4	Core damage source term release timing and aerosol thermal hydraulic input methodology changes.
4	Section 1.2 and Appendix B	Methodology for calculating environmental qualification doses in the CNV and bioshield envelope regions added. eRAI No. 9690 Question 01.05-39 Response.
4	Section 3.2.6	Clarification of iodine spike design basis source term. eRAI No. 9690, Question 01.05-41 Response.
4	Section 3.3.1	Core radionuclide inventory methodology change.
4	Section 4.3	Clarifications of aerosol removal and transport methodology.
4	Section 4.4	Added reference for cesium hydroxide descriptions.
4	Section 7.1	Updated Revision number of referenced "Quality Assurance Program Description for the NuScale Power Plant".

CONTENTS

TABLES

FIGURES

Abstract

This NuScale topical report describes the methodology used for establishing the source terms and radiological consequences for a spectrum of 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.

A methodology for establishing the NuScale iodine spike design basis source term (DBST) and core damage source term (CDST), which together meet the intent of 10 CFR 52.47(a)(2)(iv), is presented in this report. The CDST 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 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 CDST 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 Power, LLC (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 use of an iodine spike design basis source term (DBST) and a core damage source term (CDST) that together 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 Nuclear Energy Institute (NEI) position paper on small modular reactor (SMR) source terms (Reference 7.2.20), NuScale refers to 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 a facility's design features and site characteristics provide an acceptable level of protection to the public and control room operators.

10 CFR 52.47(a)(2)(iv) 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. The NuScale design has design-basis events (DBEs) that result in primary coolant entering the containment, and the iodine spike DBST described in Section 3.2.6 is used to bound the radiological consequences of these events. A beyond-design-basis core damage event (CDE) described in Section 4.2, with an associated CDST composed of a set of key parameters derived from a spectrum of accident sequences, is also postulated. The design-basis iodine

spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.47(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility's design features and site characteristics provide adequate protection of the public.

Calculations associated with the radiological consequences of the CDE 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 CDE analysis associated with the post-accident NuScale containment conditions. Consistent with RG 1.183, NuScale utilizes the assumption that no 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 CDE 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 CDE analysis.

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 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 accidents. 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 accidents. NuScale seeks approval for the methodology for establishing the iodine spike design basis source term (DBST) and the beyond-design-basis core damage source term (CDST). The iodine spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met. NuScale also seeks approval of the beyond-design-basis core damage event (CDE) 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 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 CDST 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 with regulatory acceptance criteria are provided for illustration to aid the reader's understanding of the context of the application of these methodologies.

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

1. Treatment of the CDE, postulated as a major accident for purposes of site analysis pursuant to Footnote 3 of 10 CFR 52.47(a)(2)(iv), as a beyond-design-basis event for the NuScale design.
2. Use of ARCON96 methodology for the calculation of offsite atmospheric dispersion factors.
3. {{

   }}2(a),(c)
4. Release timing values associated with the surrogate accident scenario with the minimum time to core damage are taken as the CDST release timing values.

5. Representative (median) release fractions from fuel into containment from the spectrum of surrogate accident scenarios are taken as the CDST release fractions.
6. Use of SAND2011-0128 radionuclide groups for the CDST.
7. STARNAUA is appropriate for modeling natural removal of containment aerosols for the NuScale design.
8. Utilizing thermal-hydraulic data associated with the surrogate accident scenario with the minimum time to core damage is appropriate for use in STARNAUA.
9. No maximum limit on iodine decontamination factor for natural removal of containment aerosols.
10. {{ }}2(a),(c)
11. Utilizing the iodine spiking assumptions of RG 1.183 is appropriate.
12. Utilizing the iodine decontamination factor assumptions of RG 1.183 for the fuel handling accident is appropriate.
13. With respect to accident analysis, it is appropriate to neglect the small secondary side volume that could contain activity from primary to secondary leakage for the NuScale design.
14. For pH_T values of 6.0 or greater, the amount of iodine re-evolution that could occur between pH_T values of 6.0 and 7.0 is negligible and not included in the dose calculation.
15. Containment shine of the radiation in the containment airspace through the containment vessel, reactor pool water, and then through the reactor building walls or ceiling to the environment is negligible for the NuScale design.
16. Use of the methodology for calculating environmental qualification doses in the containment vessel (CNV) and bioshield envelope regions described in Appendix B of this report.

1.3 Abbreviations

Table 1-1. Abbreviations

Term	Definition
ALWR	advanced light water reactor
AST	alternative source term
Bq	Becquerel (unit of radioactivity)
CDE	core damage event
CDST	core damage source term
Ci	curie (unit of radioactive decay)

Term	Definition
μCi	microcurie (1.0E-06 Ci) (unit of radioactive decay)
cfm	cubic feet per minute (unit of flow)
CNV	containment vessel
COL	combined license
CR	control room
CVCS	chemical and volume control system
DBE	design basis event
DBST	design basis source term
DCF	dose conversion factor
DHRS	decay heat removal system
DSRS	Design-Specific Review Standard
EAB	exclusion area boundary
ECCS	emergency core cooling system
ERF	emergency response facility
ESP	early site permit
FGR	Federal Guidance Report
FHA	fuel handling accident
GDC	General Design Criteria
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

Term	Definition
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
NEI	Nuclear Energy Institute
NPM	NuScale Power Module
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
PRCC	partial rank correlation coefficient
PWR	pressurized water reactor
RCS	reactor coolant system
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
SRP	Standard Review Plan
Sv	sievert (unit of radiation dose)

Term	Definition
TEDE	total effective dose equivalent
TMI	Three Mile Island
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.
Core damage event	A special event (beyond design basis) with radionuclides from core damage released into an intact containment postulated as the maximum hypothetical accident to enable deterministic evaluation of the response of a facility's engineered safety features.
Core damage source term	The source term associated with the core damage event that is composed of a set of key parameters derived from a spectrum of surrogate accident scenarios.
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	Radionuclide release associated with a design basis accident.
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.
Iodine spike design basis source term	The bounding source term associated with design-basis events that result in primary coolant entering the containment postulated to enable deterministic evaluation of the response of a facility's engineered safety features.
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).

Term	Definition
Single release phase	A single release phase of fission products, as opposed to distinct gap release and early in-vessel release phases.
Surrogate accident scenario	A postulated event that results in core damage with subsequent release of appreciable quantities of fission products into an intact containment, that serves as a surrogate to the large break loss-of-coolant accident with a substantial meltdown of the core typically evaluated by light water reactors as the maximum hypothetical accident.

2.0 Background

NuScale Power, LLC (hereafter, "NuScale") distinguishes between design-basis source terms (DBSTs) and the beyond-design-basis core damage source term (CDST).

The design-basis source terms include standard deterministic design-basis accidents that are similar to those of large light water reactors (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 DBST methodology is consistent with Regulatory Guide (RG) 1.183 methodology. NuScale follows the guidance of RG 1.183 for its approach to DBSTs, 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.

A NuScale-unique iodine spike DBST and CDST together meet the intent of 10 CFR 52.47(a)(2)(iv) (Reference 7.2.1). 10 CFR 52.47(a)(2)(iv) 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 Nuclear Energy Institute (NEI) position paper on small modular reactor (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. The NuScale design has design-basis events (DBEs) that result in primary coolant entering an intact containment and the iodine spike DBST described in Section 3.2.6 is used to bound the radiological consequences of these events.

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

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

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.” Therefore, a beyond-design-basis core damage event (CDE) described in Section 4.2, with an associated CDST composed of a set of key parameters derived from a spectrum of surrogate accident scenarios, is also postulated. The design-basis iodine spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design-basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.47(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility’s design features and site characteristics provide adequate protection of the public.

The methodology evaluates other design basis accidents for radiological consequences. Table 2-1 is a summary of applicable events for the NuScale design. The table includes the event, the NuScale DSRS or SRP section that addresses the event, the section of RG 1.183 that discusses the event input parameters and assumptions, and the primary source of radiation for the event.

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

Event	DSRS or SRP	RG 1.183 Appendix*	Primary Source of Radiation
Core damage event	15.0.3	A	damaged fuel
Fuel handling accident	15.7.4	B	
Rod ejection accident	15.4.8	H	
Main steam line break	15.1.5	E	coolant activity (with iodine spiking)
Steam generator tube failure	15.6.3	F	
Primary coolant line break	15.6.2	n/a	
Iodine spike DBST**	15.6.5	n/a	

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

**Note: The iodine spike DBST is not an event, but rather a bounding source term associated with DBEs that result in primary coolant entering the containment.

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.

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.

See Appendix A for a summary of how pertinent requirements are met without a design-basis core damage event. See the NuScale Accident Source Term Regulatory Framework white paper (Reference 7.1.4) for additional background information.

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
- RG 1.195, "Methods and Assumptions for Evaluating Radiological Consequences of Design Basis Accidents at Light-Water Nuclear Power Reactors," Revision 0, May 2003
- SECY-94-302, “Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light-Water-Reactor Designs,” December 1994

2.2 Basis for Treating Core Damage Event as a Beyond Design Basis Event

NuScale treats the core damage event as a beyond-design-basis event. This approach is based on attributes of the NuScale design as compared to the history of the 10 CFR 52.47(a)(2)(iv) dose evaluation requirement and related developments in NRC policy and guidance. These developments, as summarized below, indicate it can be acceptable to consistently treat a core damage event, which is a severe accident, as a beyond-design-basis event for a design such as NuScale's.

In SECY-03-0047 (Reference 7.2.16), staff recognized that the classic LWR siting evaluation based on an in-vessel core melt may not be applicable to non-LWR designs. Staff recommended the use of “scenario-specific source terms” derived from DBEs defined for the plant, allowing “credit to be given for unique aspects of plant design” (i.e., performance-based). SECY-05-0006 (Reference 7.2.17) discussed broadening that same framework to include future LWRs.

In SECY-90-016 (Reference 7.2.28), staff stated “severe core damage accidents should [not] be design basis accidents (DBA) in the traditional sense that DBAs have been treated in the past,” in that features provided for only severe accident mitigation were not required to meet typical safety-related requirements. Because the MHA is a severe core damage accident, its treatment as a design basis accident for the purposes of evaluating offsite doses pursuant to 10 CFR 52.47(a)(2)(iv) limits implementation of the NRC position on equipment survivability established by SECY-90-016 and the associated Staff Requirements Memorandum.

In 10 CFR 50.67 rulemaking (Reference 7.2.29), the NRC stated that “there is no regulatory requirement for a specific source term for reactors to be licensed in the future.”

In the federal register notice associated with the 10 CFR Part 100 rulemaking (Reference 7.2.32), the NRC stated “It is worth noting that events having the very low likelihood of about 10^{-6} per reactor year or lower have been regarded in past licensing actions to be ‘incredible’, and as such, have not been required to be incorporated into the design basis of the plant.”

A scenario resulting in core damage for the NuScale Power Module (NPM) design would entail multiple failures of safety-related equipment. No electrical power or operator action is required to prevent or mitigate a design-basis accident. Passive fail-safe safety systems, a small core size, and a CNV that is partially submerged in the ultimate heat sink result in the frequency of a potential core damage in an NPM being orders of magnitude lower than the safety goals, and core damage sequences with frequencies that are orders of magnitude lower than those considered "incredible" in past licensing actions. Dominant NPM risk contributors involve one or more common cause failures of redundant components, such as 3 of 3 emergency core cooling system (ECCS) reactor vent valves or 4 of 4 decay heat removal system actuation valves, and common cause initiating events, such as failure of 4 of 4 DC buses. From these observations, it follows that a severe accident scenario resulting in core melt for the NuScale design constitutes a beyond-design-basis event, and the consideration and mitigation of dose consequences for such an event should be treated accordingly.

3.0 Methodology Overview

This topical report presents the methodology utilized to perform the radiological calculations associated with the DBSTs and CDST, with a focus on the following NuScale specific methodology:

- iodine spike DBST
- atmospheric dispersion
- core damage source term
- containment aerosol generation and removal
- post-accident pH_T

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.

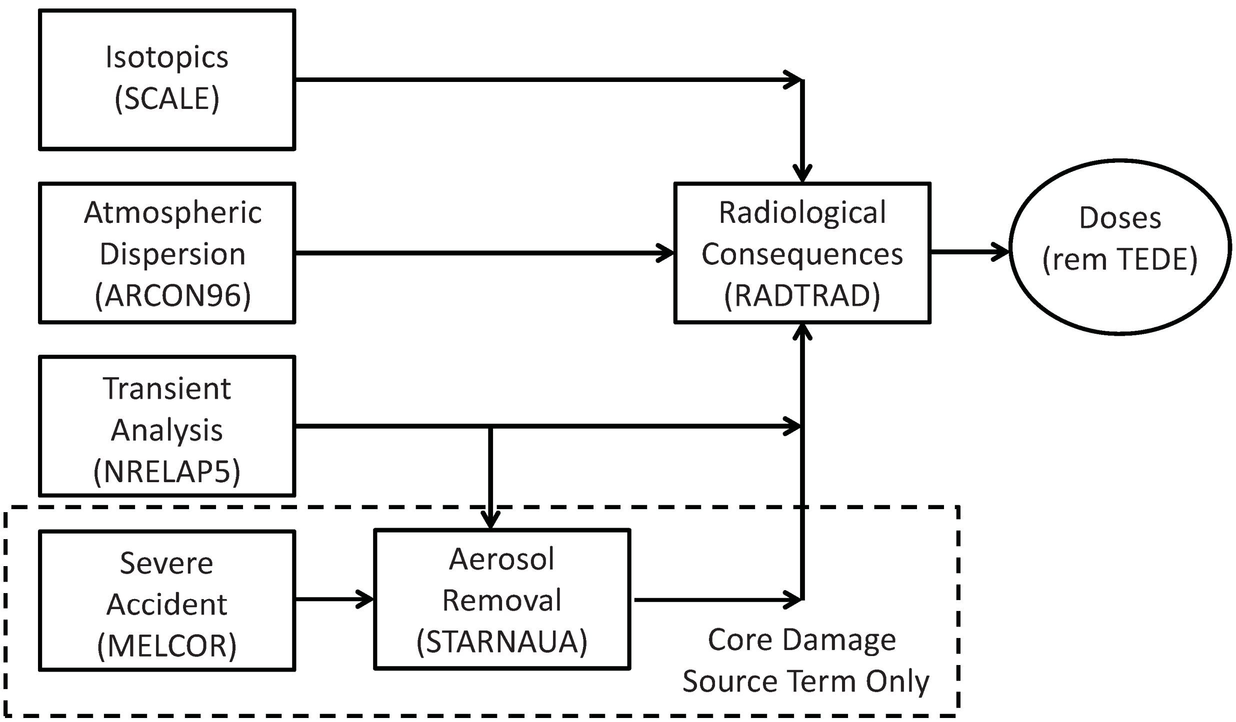

Figure 3-1. Flowchart of accident radiological calculation process

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 NARCON

The calculation of both onsite and offsite atmospheric dispersion factors for design basis accidents and the CDE is performed with NARCON. NARCON is the NuScale version of ARCON96 (Reference 7.2.24). NARCON is equivalent to ARCON96 with the exceptions of input/output edit differences {{

}}2(a),(c)

ARCON96 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 design-basis accidents and the CDE (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 the CDST 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
- 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 was 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 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 sedimentation 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 4.4. 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 MCNP6

MCNP6 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 Design Basis Source Terms

3.2.1 Rod Ejection Accident

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. For the purposes of this report, the REA methodology is described assuming the results from the failure of one full assembly in order to establish a methodology in the event that fuel failures are postulated in a future application. If fuel failure does occur, then the radiological consequence analysis considers the REA event with two different release paths, as described below.

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 percent cesium iodide, 4.85 percent elemental iodine, and 0.15 percent organic iodide
- primary system 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
- 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, as updated by Regulatory Issue Summary (RIS) 2006-04 (Reference 7.2.11) item 8, is followed. The methodology assumes failure of all the fuel rods in one irradiated fuel assembly 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.

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
- overall effective iodine decontamination factor of 200 for the pool
- 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.

This radiological consequence analysis considers the chemical and volume control system (CVCS) break event with a coincident iodine spike. The following is a description of the postulated scenario evaluated for this event.

1. A failure of CVCS piping occurs outside the containment vessel inside the reactor building.
2. The primary coolant contains iodine and noble gas radionuclides.
   a. Coincident iodine spike: The primary coolant inside the reactor vessel initially contains no iodine. Over a period of eight hours, iodine is transported from inside the fuel rods to the coolant at a rate 500 times the normal release rate. Coolant inside the CVCS equipment and piping is at the maximum concentrations allowed by design basis limits.
   b. Noble gas: The primary coolant contains the maximum concentrations allowed by design basis limits.
3. Before containment isolation occurs, primary coolant flows out the reactor vessel through the break at a rate and duration calculated by the transient analysis. This

results in a time-dependent release of activity to the reactor building which is modeled for conservatism as a direct release to the environment through the break.

4. After containment isolation, primary coolant leaks through one containment isolation valve (other in-series valve is assumed to fail) at the maximum leak rate allowed by design basis limits. The leakage continues until the reactor is brought to shutdown conditions. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by any intervening structures.
5. Available primary coolant in the CVCS equipment (heat exchangers, filters, etc.) and piping flows out of one or the other side of the break. The coolant is at the maximum activity concentration allowed by design basis limits.
6. Once the reactor is completely shut down and depressurized, all releases through valve leakage stops.

The following is a summary of the explicit assumptions used from Appendix E (main steam line break) of RG 1.183.

- coincident iodine spiking factor: 500
- duration of coincident iodine spike: 8 hr
- 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

3.2.6 Iodine Spike Design Basis Source Term

The iodine spike DBST is composed of a set of key parameters, derived from the assumption of a generic failure occurring inside the CNV, which results in the release of all primary coolant from the reactor coolant system (RCS) to the CNV. The iodine spike DBST is a surrogate that bounds the radiological consequences of a spectrum of events that result in primary coolant entering an intact containment.

Primary coolant with radionuclide concentrations at the design basis limits enters the containment and 100 percent of the radionuclides within 100 percent of the primary coolant are assumed to be present in the containment. This assumption is conservative because some amount of primary coolant (at least the amount required to cover the core) would remain in the reactor pressure vessel (RPV) and, therefore, the radionuclides associated with that primary coolant would not be available in the CNV for release. Additionally, this is conservative because some amount of the radionuclides would remain in the primary coolant at the bottom of the CNV, but the analysis assumes all the radionuclides are available to leak out of the CNV as vapor. Because the iodine spike DBST is not a specific event, nor an extension of a specific event, there is no thermal-hydraulic analysis associated with the iodine spike DBST.

This radiological consequence analysis considers the iodine spike DBST with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. These iodine spikes are derived as shown in

Section 3.3.2 of this report. A description of the evaluated scenario is summarized as follows:

1. A generic failure is assumed to occur inside the CNV, resulting in the release of all primary coolant from the RCS to the CNV.
2. The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios.
3. Primary coolant flows into the CNV through a nonspecific release point with an instantaneous release of activity into the CNV. The release is homogenously mixed as vapor throughout the entire CNV free volume.
4. Activity is then assumed to leak into the environment at the design basis leakage rate for 24 hours, then at 50 percent of the design basis leakage rate thereafter. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures. Aerosol removal is not credited.
5. Once the reactor is completely shut down and depressurized, all releases through the containment to the environment stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- Coincident iodine spiking factor – 500 (because this is the largest coincident iodine spiking factor recommended for any event in RG 1.183)
- Duration of coincident iodine spike – 8 hours
- Iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide (arbitrary assumption because RADTRAD requires the input, but this assumption has no impact on results)
- Activity released from the fuel due to the pre-incident iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the CNV; activity released from the fuel due to the coincident iodine spike is assumed to mix instantaneously and homogeneously within the fuel volume, then release to the CNV over the 8 hour coincident spiking duration
- No reduction or mitigation of noble gas radionuclides released from the primary system

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 at end of cycle 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	62 GWd/MT
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 ($A_i^{\mathrm{Re}leased}$) is determined by the release fractions defined for the event and isotope, expressed as

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

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 (62 GWd/MT 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. Actual isotopic concentrations are derived from the nominal coolant concentrations calculated and then scaled to the design basis limit maximum. In addition, a release rate factor from the fuel for the coincident iodine spike scenario is applied as specified in SRP Section 15.6.2.

With appropriate unit conversions, concentration of each isotope may be calculated by Equation 3-2.

$$\left(\frac{\mu Ci}{gm}\right)_i = \frac{(Ci)_i}{V\ ft^3} \cdot \frac{ft^3}{\rho\ lbm} \cdot \frac{lbm}{453.59gm} \cdot \frac{\mu Ci}{10^{-6}Ci} \qquad \text{Eq 3-2}$$

where:

$(Ci)_i$ = Total activity for isotope i

$\left(\frac{\mu Ci}{gm}\right)_i$ = Concentration for isotope i

V = Volume of primary coolant, ft^3

ρ = Density of primary coolant, lbm/ft^3

The isotopic concentrations equivalent to the design basis limit DE I-131 is determined from the nominal concentration using dose conversion factors from Reference 7.2.38 as described by Equation 3-3.

$$\left(\frac{uCi}{gm}\right)_i^{DE} = \left(\frac{uCi}{gm}\right)_i^{Nom} \times \frac{DCF_i}{DCF_{I-131}} \qquad \text{Eq 3-3}$$

where:

i = index for isotopes

DCF_i: = Dose Conversion Factor for isotope i

DCF_{I-131} = Dose Conversion Factor for isotope I-131

$\left(\frac{uCi}{gm}\right)_i^{DE}$ = Concentration of isotope i adjusted for DE I-131

$\left(\frac{uCi}{gm}\right)_i^{Nom}$ = Nominal concentration of isotope i

For the pre-incident iodine spike scenario, the same technique is utilized to calculate the primary coolant design basis limit maximum source terms for a dose equivalent I-131 at the elevated pre-incident limit. In both spiking scenarios, the noble gas concentrations are calculated using the same technique scaled to the design basis limit for DE Xe-133 and the dose conversion factors from Reference 7.2.39 as defined by

$$\left(\frac{uCi}{gm}\right)_i^{DE} = \left(\frac{uCi}{gm}\right)_i^{Nom} \times \frac{DCF_i}{DCF_{Xe-133}} \qquad \text{Eq 3-4}$$

In order to model a coincident iodine spike as specified in RG 1.183, the primary coolant iodine concentration is estimated using a “spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value [event-specific] times greater than the release rate corresponding to the iodine concentration at the equilibrium value” (Reference 7.2.2).

At equilibrium the production and removal of iodine is equal, by definition. Therefore, the equilibrium iodine release rate, (i.e., iodine production rate) may be calculated by determining the iodine removal rate. The removal rate is a function of the reactor coolant system (RCS) removal, CVCS decontamination, and the natural radioactive decay process. For conservatism, infinite decontamination of the CVCS is credited, as doing so would result in a higher production rate. Therefore, for each isotope i, the isotopic production or removal rate λ_i is the sum of the two removal mechanisms of RCS removal $\lambda_{i,RCS}$ and radioactive decay $\lambda_{i,decay}$, expressed algebraically in the form of

$$\lambda_{i,Production} = \lambda_{i,Removal} = \lambda_{i,RCS} + \lambda_{i,decay} \qquad \text{Eq 3-5}$$

where, $\lambda_{i,RCS}$ is derived from contributions of CVCS letdown flow and allowable RCS operational leakage.

The equilibrium production rate (R_i) is then the activity of each isotope (A_i) multiplied by the appearance rate of the isotope, as in

$$R_i = A_i \times (\lambda_{i,RCS} + \lambda_{i,decay}) \qquad \text{Eq 3-6}$$

For all isotopes, the removal rate of iodine by RCS removal is based on the flow rate and the total mass of coolant (M) in the primary system. Applying the appropriate unit conversions results in an expression of the form

$$\lambda_{i,RCS}(sec^{-1}) = \frac{Flow\ gal}{min} \cdot \frac{1}{M\ lbm} \cdot \frac{\rho\ lbm}{ft^3} \cdot \frac{ft^3}{7.481\ gal} \cdot \frac{min}{60\ sec} \qquad \text{Eq 3-7}$$

The decay rate of each isotope is determined from the half-lives $T_{½}$ of each isotope, using

$$\lambda_{i,decay} = \frac{\ln|2|}{T_{½}} \qquad \text{Eq 3-8}$$

Applying the above equations to each isotope of iodine results in the production rates and total activities at the end of the 8 hour spiking period. The release rate from the fuel to primary coolant is calculated such that 99 percent of the activity available for release from the fuel is released to the coolant in the eight-hour duration of the concurrent iodine spike utilizing an exponential release model.

$$A = A_o e^{-\lambda t} \qquad \text{Eq 3-9}$$

where,

A_O = Initial Activity in Fuel

A = Final Activity in Fuel

λ = Removal Rate (1/hr)

t = Removal Time (hr) = 8.0 hr

Solving Eq 3-9 for the removal rate and including an arbitrary fuel volume of 500 ft^3. The volume of the fuel is arbitrary, given that 99 percent of the activity will be released irrespective of the fuel volume. The reason for the assumption of 99 percent is that as an exponential release model, 100 percent will never be reached. The arbitrary fuel volume is used to convert the release rates into units of cubic feet per minute (cfm), which is the only input allowed by RADTRAD. Converting into units of cfm results in

$$\frac{\ln(1\%)}{-8hr}\cdot 500ft^3\cdot\frac{hr}{60\min}=4.8cfm \qquad \text{Eq 3-10}$$

The total activity in the fuel eligible to be released from the fuel into the coolant due to spiking C_i is based on the event-specific spiking factor *SF* multiplied by the equilibrium production rate R_i, as in

$$C_i = SF \cdot R_i \qquad \text{Eq 3-11}$$

Thus, the iodine spike can be modeled as a “flow rate” from the fuel to the coolant for the eight-hour duration using Eq 3-10, with the total activity for each isotope in the fuel eligible for release calculated from Eq 3-11.

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. The ratio of the secondary coolant that could contain iodine to the primary coolant is small (approximately 1 percent) and therefore the potential source of radioactivity in the secondary coolant that is typically accounted for in PWR dose analysis is neglected in NuScale’s design.

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 application that references this report.

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

The final control room ventilation design is expected to be provided in the application that references this report. A representative design for the control room ventilation system was used to confirm the methodology assumptions. The key design features assumed for this representative example design are summarized as follows:

- The nonsafety related normal control room ventilation is isolated by a control system once a sufficiently high source of radioactivity is measured
- 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	1
Control Room Envelope Volume	ft^3	170,000
Control Room Emergency Flow Rate	cfm	100
Control Room Normal Flow Rate	cfm	495
Control Room Emergency Duration	hr	72
Control Room Unfiltered Ingress/Egress	cfm	5
Control Room Unfiltered In-leakage	cfm	10

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

The control room exhaust is equal to the total inlet flow in both the normal and emergency operating modes. For the pressure to remain in equilibrium, an equal amount of air must be exhausted. This assumption takes credit for air escaping through doors when open or other potential leakage pathways, such as penetrations in the control room envelope.

3.3.4.6 Radiation Shine Radiological Consequences

Per Section 4.2.1 of RG 1.183 (Reference 7.2.2), the following contributions of radiation shine to the control room dose are included in the methodology:

- radioactive material in systems and components inside or external to the control room envelope, for example, radionuclides collected in filters
- sky shine from an external radioactive plume released from the facility
- direct shine from airborne fission product gases within the reactor building and contamination of structural surfaces within the reactor building

The calculated shine dose from the bounding event is applied to all events rather than specifically calculating the shine dose for each event. For the event that results in the largest activity released into the reactor building and control room, the radionuclide activities calculated by RADTRAD at key time intervals throughout the event are utilized as input into ORIGEN-S and MCNP models.

The photon source terms are defined according to a user-supplied energy group structure, which consists of a series of energy bins over which average source strengths are calculated. In the ORIGEN-S calculations, an eighteen-group energy structure is adopted for the photon spectra.

MCNP calculations include explicit models of the control room and surrounding reactor building in conjunction with a defined radiation source. The source configurations considered include airborne fission product gases within the environs of the reactor building, radionuclide contamination of structural surfaces within the reactor building, and radionuclide contamination within the control room external filter.

In each calculation model, the exposure to an operator within the control room is evaluated using the cell (volume) tally feature of the MCNP code. The MCNP calculations employ the continuous energy cross section data libraries from the associated ENDF/B-VII nuclear data libraries and employ photon flux-to-dose conversion factors that are used to evaluate the exposure to an operator within the control room as a function of the operator's position. The photon flux-to-dose conversion factors employed are based on International Commission on Radiological Protection (ICRP)-74 (Reference 7.2.61).

Shine through the containment vessel, reactor pool water, and then through the reactor building walls or ceiling to the environment is assumed to be negligible for the NuScale design. More than half of the containment vessel is assumed to be submerged in the reactor pool for the majority of the thirty day evaluation period. This water would provide significant shielding, with further shielding provided by multiple feet of reinforced concrete walls and the concrete roof of the aircraft impact resistant reactor building.

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.

The methodology assumes that the reactor pool water would contain radionuclides primarily from routine refueling operations of a number of modules over a period of time. A lifetime equilibrium tritium concentration is assumed based on estimated boron concentrations in the reactor coolant and radwaste recycling associated with the pool cleanup system. Isotopes considered are those isotopes deemed important contributors to dose, and that are predicted to occur in the reactor pool, specifically the isotopes of H-3, I-131, Cs-137 and Sr-90.

The radioactive source term for this event are the nuclides released in the reactor pool water from refueling operations. These values are multiplied by the cumulative pool water volume released during boil off. {{

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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

The methodology presented in this topical report assumes that there is no direct release path from the reactor building or turbine generator building to the control room or technical support center. This hypothetical flow path would bypass the environment and would not consider atmospheric dispersion.

The methodology assumed an example design with a connection between the control building and reactor building. In this example a tunnel connects the reactor building to the control building. Between the reactor building and control room envelope, it is assumed that there are multiple doors, multiple airlocks, and administrative controls on ingress and egress through these doors. The two turbine generator buildings are assumed physically separate from the control building with no interconnecting passageway. The validity of the assumption that radiation could not credibly be transported through this pathway in a postulated design basis accident or the CDE is expected to be confirmed by the application that references this topical report.

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

For both DBST and CDST radiological consequence analyses, with the exception of the failure of small lines carrying primary coolant outside containment analysis, the containment is assumed to leak at the design basis limit leak rate for 24 hours and then at half of the design basis limit leak rate thereafter. As described in Section 3.2.5 of this report, the failure of small lines carrying primary coolant outside containment analysis conservatively assumes the maximum leak rate allowed by design basis limits until the reactor is shut down and depressurized, even if that time is beyond 24 hours.

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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. An elemental decontamination factor of 285, an organic decontamination factor of 1, and an overall effective decontamination factor of 200 are assumed per RG 1.183 as updated by RIS 2006-04 (Reference 7.2.11) item 8.

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.

NuScale intends to follow the current regulatory guidance with respect to modeling iodine spiking as the justifications made in GI-197 are applicable to NuScale fuel and reactor coolant system operating conditions.

GI-197 discusses the adequacy of current industry practice iodine spiking modeling in detail, focusing on SGTF for a large PWR plant in which it is assumed that isolation will take up to two hours. GI-197 also discusses that iodine spiking assumed to occur over 8 hours may be approximated by a more severe two hour spike. Thus, for this type of event in which the spiking duration is proportional with the time required for isolation, NuScale utilizes the iodine spiking assumptions of RG 1.183.

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 anticipates the possibility of applicants referencing this topical report postulating 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 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.

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

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

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

$\frac{X}{Q}(x,i,j)$	= 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

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

[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 overall 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

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

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

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. {{ }}2(a),(c)
- 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. NuScale's methodology is to utilize {{

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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. {{

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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.

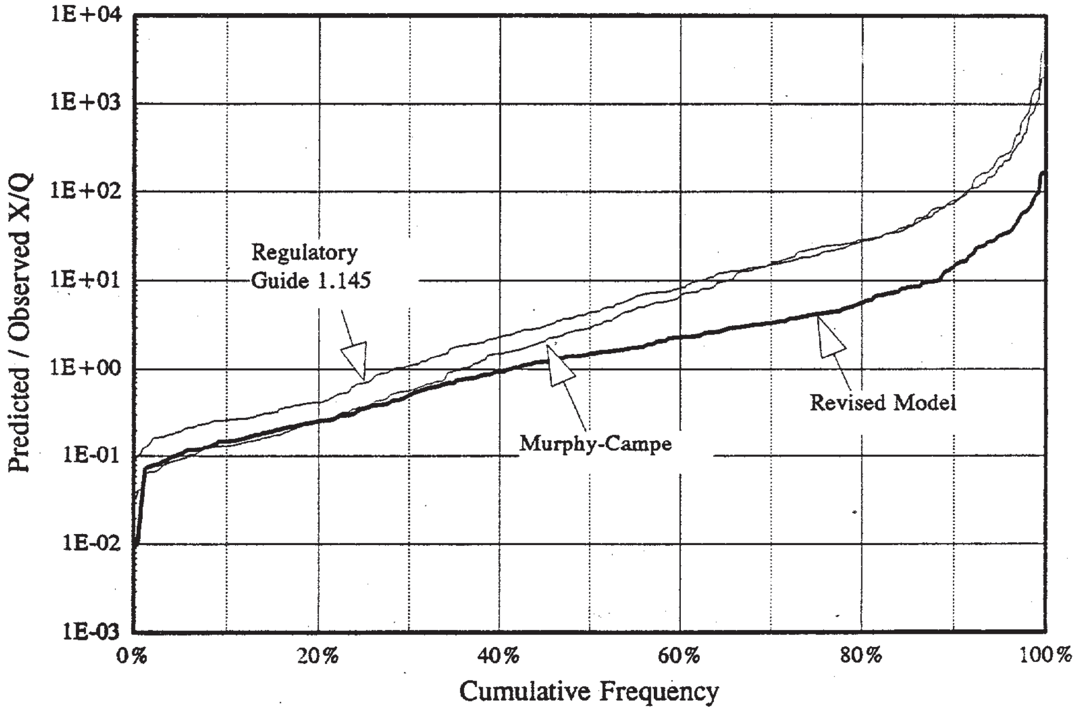

Figure 4-1. Cumulative frequency distributions of predicted to observed concentration ratios for the Murphy-Campe (RG 1.145), and revised models (Reference 7.2.23)

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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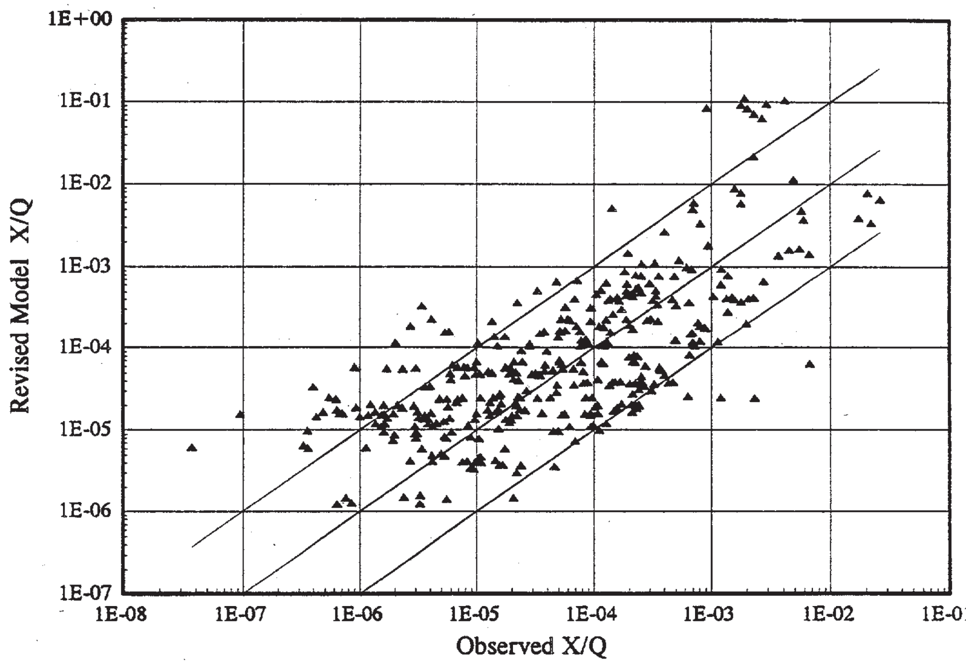

Figure 4-4. Comparison of ARCON96 concentration predictions with observed values (Reference 7.2.23)

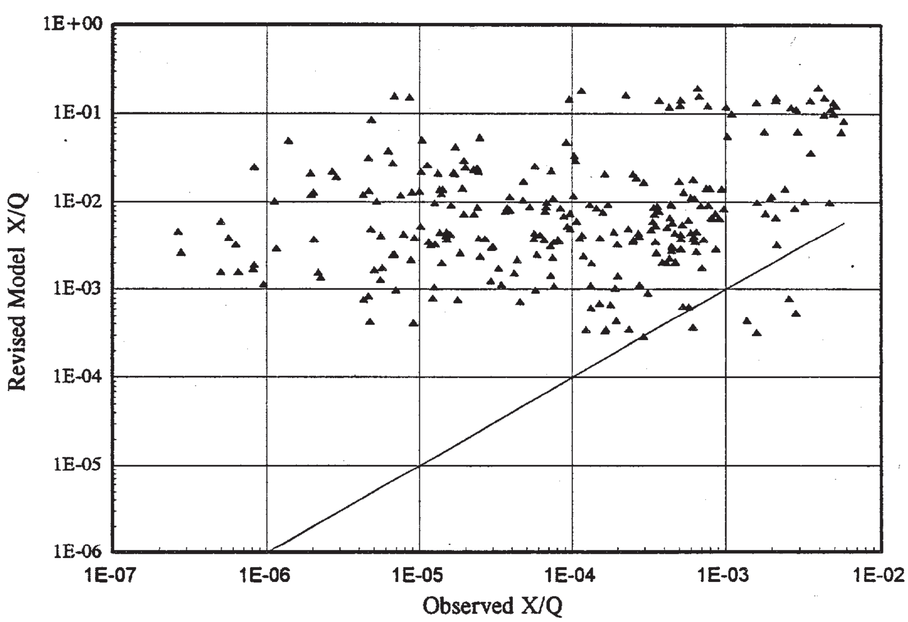

Figure 4-5. Comparison of ARCON96 concentration estimates with observed values in the building surface data set (Reference 7.2.23)

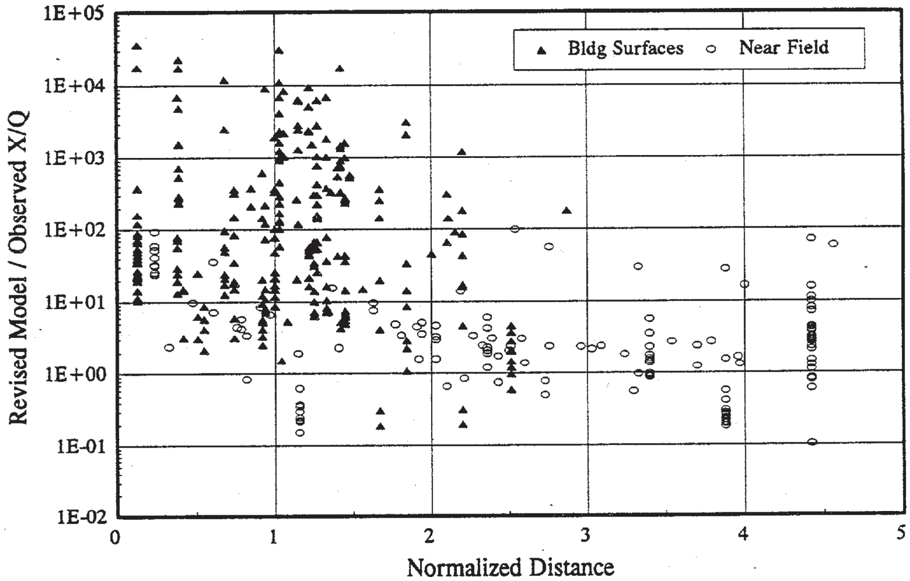

Figure 4-6. Ratios of predicted to observed concentrations for ARCON96 (Reference 7.2.23)

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Figure 4-7 and Figure 4-8, also taken from Reference 7.2.24, {{

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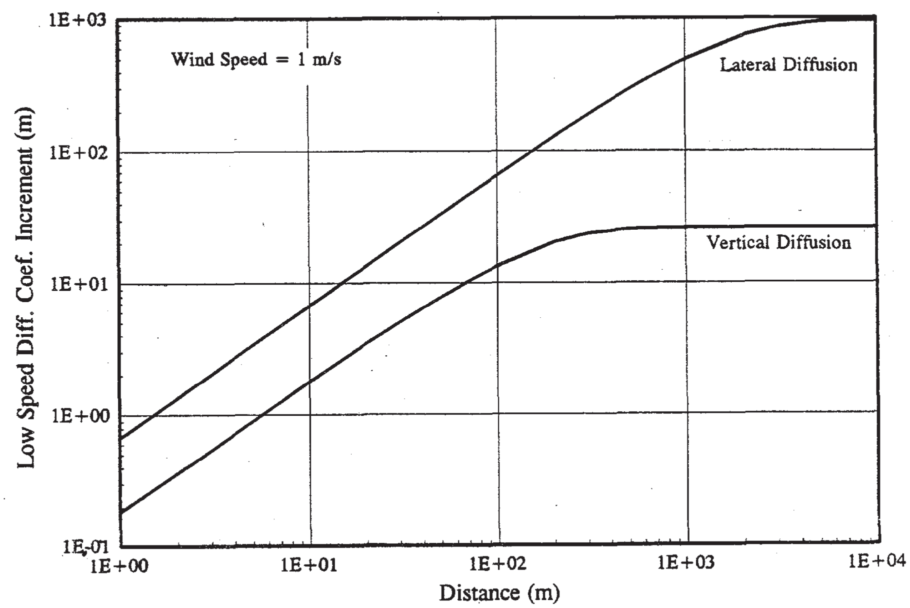

Figure 4-7. Variation of low speed diffusion coefficient increments as function of distance (Reference 7.2.23)

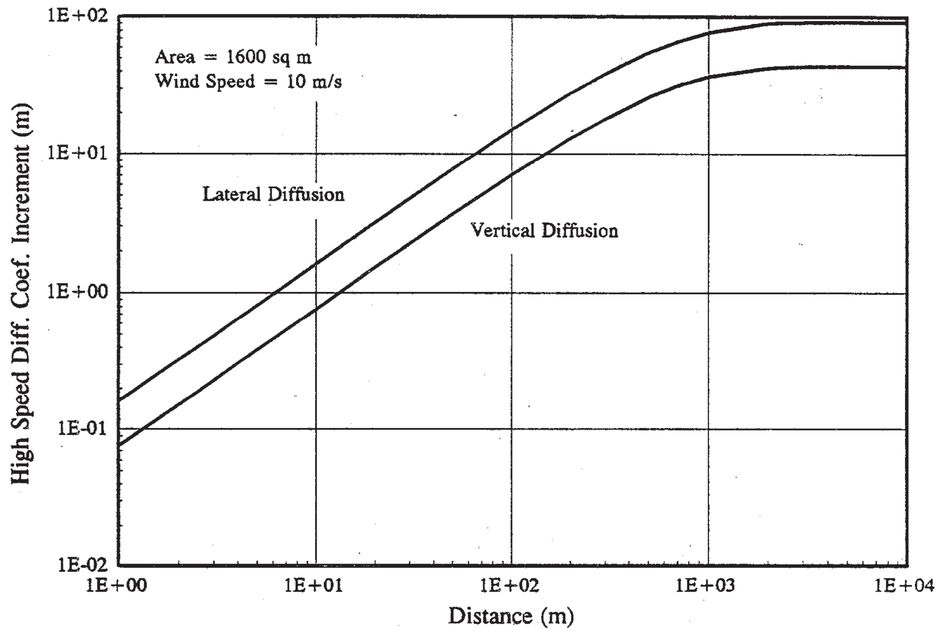

Figure 4-8. Variation of high speed diffusion coefficient Increments as function of distance (Reference 7.2.23)

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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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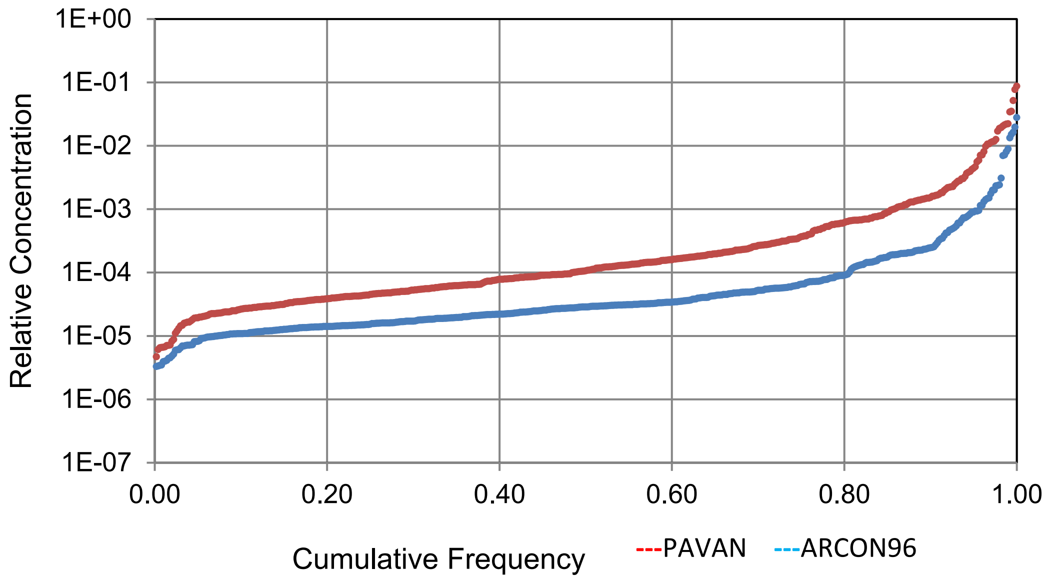

Figure 4-9. Cumulative frequency distributions of predicted concentrations for PAVAN and ARCON96 methodologies

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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.

- For each possible measured wind direction sector available in the input meteorological data (typically 16 sectors), {{

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- Ground level release (no credit taken for possible elevated release)
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4.2 Core Damage Event

NuScale postulates a CDE to provide reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility's design features and site characteristics provide adequate protection of the public and operators. The CDE associated CDST is composed of a set of key parameters, such as fuel release fractions and timing, derived from a spectrum of surrogate accident scenarios that are utilized as inputs in radiological consequence calculations associated with the MHA.

The CDST is based on a major accident, postulated for the purpose of design analyses. Events that involve core damage with the subsequent release of appreciable quantities of fission products into an intact containment, as described in 10 CFR 52.47(a)(2)(iv), are addressed in the methodology.

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 CDE radiological consequence calculation.

4.2.1 Definition of Core Damage Source Term

A subset of the Level 1 PRA sequences is used to select the spectrum of surrogate accident scenarios considered for the establishment of the CDST. These sequences are all single module internal events at full power and assume an intact containment. The methodology directs that a range of surrogate accident scenarios be 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 surrogate accident scenarios. For each surrogate accident scenario, 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 release duration associated with minimum release onset time, and the median value of the release fractions determined from the spectrum of surrogate accident scenarios are established and used as the CDST.

A summary of the CDST 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
- leakage from containment is direct to the environment, with no removal mechanisms in reactor building (e.g., scrubbing, partitioning, deposition)
- 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 CDE onsite and offsite radiological consequence estimates utilize 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 considers cesium iodide as an aerosol.

It is conservative for the CDE onsite and offsite radiological consequence estimates to follow the general methodology described in Section 3.0 of this report. Because the CDE is a beyond-design-basis event, more realistic analysis techniques may be utilized with appropriate justification. For example, with sufficient justification provided by the application referencing this report, it may be appropriate to utilize nominal atmospheric dispersion factors and isotopic inventories instead of 95th percentile or bounding values, or to credit deposition of radionuclides within the reactor building.

4.3 Aerosol Removal and Transport

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Table 4-5. Summary of aerosol parameter ranges

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Section 3.2 of Appendix A in RG 1.183 allows for credit to be taken for the reduction in airborne radioactivity in the containment by natural deposition within the containment. Specifically, two models are endorsed for use: (i) the SRP 6.5.2 model, or (ii) the NUREG/CR-6189 model. The model described in NUREG/CR-6189 (Reference 7.2.49) for natural aerosol removal applies to reactors of 1000 MWt and larger, and does not consider removal mechanisms other than sedimentation. As a result, this model does not apply to the NuScale design.

Although Section 6.5.2 of the SRP addresses active containment spray systems, which the NuScale design does not have, the guidance describes a methodology acceptable to the NRC Staff for calculating fission product removal rates. Specifically, the guidance states models that include the following characteristics are reviewed on a case-by-case basis:

- chemical and physical processes that can occur during an accident
- mass-mean diameter of water droplets
- average water droplet fall height
- area of the interior surfaces of the containment

In conformance with the aforementioned guidance from SRP 6.5.2, {{

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4.3.1 STARNAUA

As described in Section 3.1.6, STARNAUA models natural removal of containment aerosols by sedimentation and diffusiophoresis, and considers the effect of hygroscopicity (growth of hygroscopic aerosols due to steam condensation on the aerosol particles) on aerosol removal. STARNAUA also includes a model for thermophoresis and the capability to directly input steam condensation rate or condensation heat transfer rate, and total heat transfer rate (as provided externally via MELCOR containment thermal hydraulics code calculation). Tracking of particle size over time is based on a coagulation model that determines particle coagulation rate, and thus particle size distribution and settling velocities. Aerosols are discretized into a size distribution, or size bins, and tracked. The modeling of these phenomenon are described in greater detail in Sections 4.3.2-4.3.4.

4.3.2 Sedimentation

The STARNAUA sedimentation removal, or gravitational settling, model uses the same deposition velocity formula as that used in the removal model of NUREG/CR-6189. The settling velocity for sedimentation is derived in Reference 7.2.45 from the Stokes equation as follows:

$$v_s = 2\rho_p g r^2 C(r)/9\mu\chi \qquad \text{Eq 4-7}$$

where
ρ_p = particle material density
g = gravitational acceleration
r = particle radius
$C(r)$ = Cunningham slip correction factor
μ = viscosity of the atmosphere
χ = dynamic shape factor

The Cunningham factor is given by

$$C(r) = 1 + 1.246Kn + 0.42Kn \cdot \exp\left(-0.87/Kn\right) \qquad \text{Eq 4-8}$$

where Kn is the Knudsen number λ/r and λ is the molecular mean free path in the atmosphere.

4.3.3 Phoretic Phenomena (Diffusiophoresis and Thermophoresis)

Diffusiophoresis and thermophoresis are two phenomena in which aerosol molecules adhere to a surface due to a gradient, specifically steam (diffusio) or temperature (thermo). This phenomenon is not to be confused with plating in a physical chemistry sense of chemical adsorption that involves a chemical reaction between a molecule and a surface. STARNAUA phoretic phenomena modeling is consistent with aerosol removal modeling previously used in the design certification applications of passively cooled containment applications.

For diffusiophoretic deposition velocity the following equation is used (Reference 7.2.45).

$$v_{sf} = \frac{x_s\sqrt{M_s}}{x_s\sqrt{M_s} + x_a\sqrt{M_a}}\frac{W}{\rho_s} \qquad \text{Eq 4-9}$$

where
x_s = steam mole fraction in the atmosphere
x_a = air mole fraction in the atmosphere
M_s = molecular weight of steam
M_a = molecular weight of air
W = steam condensation rate per unit area on wall surface
ρ_s = steam density in the atmosphere

The thermophoresis model uses the equation of Talbot et. al. (Reference 7.2.47) for the thermophoretic deposition velocity. Reference 7.2.43 shows the equivalent thermophoretic velocity u_{th} (shown here as v_{th} for consistency) as

$$v_{th} = \frac{2C_s v_g(\alpha + C_t Kn)[C(Kn)]\left(\frac{\nabla T_g}{T_{g0}}\right)}{(1 + 3C_m Kn)(1 + 2\alpha + 2C_t Kn)} \qquad \text{Eq 4-10}$$

where $\alpha = k_g/k_p$ is the ratio of the thermal conductivities of the gas and particle. It is internally calculated at each time step. The term k_p is set equal to the thermal conductivity of water.

where
T_g is the gas temperature
T_{g0} is the gas temperature far from the surface
∇T_g is the normal derivative of the gas temperature at the surface

v_g is the kinematic gas viscosity

Kn is the Knudsen number and $C(Kn)$ is the Cunningham slip correction factor, as have been previously defined. C_s , C_t , and C_m are the slip, thermal, and momentum accommodation coefficients, respectively. Talbot et. al. give their best values as $C_s = 1.17$, $C_t = 2.18$, $C_m = 1.14$.

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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{{ }}[2(a),(c)] The model used in STARNAUA for non-hygroscopic and hygroscopic condensational growth on particles is based on the Mason equation and is provided in Reference 7.2.45 as follows.

Non-hygroscopic condensational growth:

$$r\frac{dr}{dt} = \frac{S - exp\left(\frac{2\sigma M_w}{\rho_w RTr}\right)}{\frac{L\rho_w}{kT}\left(\frac{LM_w}{RT} - 1\right) + \frac{\rho_w RT}{M_w D p_{sat}(T)}} \qquad \text{Eq 4-11}$$

where

S	=	bulk steam saturation ratio
σ	=	surface tension of water on the particle or droplet
ρ_w	=	density of water on the droplet
L	=	latent heat of water on the droplet
k	=	heat conductivity of water vapor
M_w	=	molecular weight of water
D	=	binary diffusion coefficient of the steam/air atmosphere
p_{sat}	=	steam saturation pressure
R	=	universal gas constant
T	=	absolute temperature

For hygroscopic particles, the above equation must be modified as follows.

Hygroscopic condensational growth:

$$r\frac{dr}{dt} = \frac{S - Aexp\left(\frac{2\sigma M_w}{\rho_w RTr}\right)}{\frac{L\rho_w}{kT}\left(\frac{LM_w}{RT} - 1\right) + \frac{\rho_w RT}{M_w D p_{sat}(T)}} \qquad \text{Eq 4-12}$$

The exponential term in the non-hygroscopic condensational growth equation represents the lowering of the saturation pressure at the surface of the droplet due to the curvature of the droplet, the so-called Kelvin effect. If the particle is hygroscopic or soluble, then there is an additional decrease in the saturation pressure due to the solute effect. This leads to an additional water activity term A in the numerator of the equation. For dilute solutions with many hygroscopic species, the water activity can be determined by the following Reference 7.2.45 equation.

$$A = \frac{1}{1 + \sum_i Q_i M_w m_i} \quad \text{Eq 4-13}$$

Where Q_i is the van't Hoff factor and m_i is the molality of each hygroscopic material dissolved in the droplet.

4.3.5 Aerosol and Elemental Iodine Removal

Deposition of aerosol onto inner-containment surfaces results in removal. A key assumption of the NuScale aerosol transport methodology is no maximum iodine decontamination factor limit should be applied to natural aerosol removal phenomenon in the NuScale containment. 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 NuScale removal rate calculation methodology is based on calculated time-dependent airborne aerosol mass. This basis 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." NuScale conservatively does not take credit for elemental iodine removal. Rather, only aerosol removal is credited.

4.3.6 Aerosol Resuspension and Revaporization

Resuspension of aerosol particles from deposition surfaces is generally considered a transport mechanism of importance only for pipes. Additionally, resuspension is based on turbulent flow conditions (Reference 7.2.43), {{

}}2(a),(c)

Revaporization is considered a negligible transport mechanism in the containment. It is noted in Reference 7.2.43 that revaporization is unlikely for fission products of interest, considering maximum post-LOCA temperatures compared against vaporization temperature/pressure curves for fission products of interest.

Although the aerosol transport modeling methodology does not include resuspension and revaporization effects explicitly, resuspension and revaporization effects are inherently included in the benchmarked experiments. {{

}}2(a),(c)

4.3.7 Charge Effects on Aerosol Removal Rates

The effect of electrical charge on aerosol transport and removal in a containment atmosphere during accident conditions has been examined by several researchers, such as in Reference 7.2.49. A tendency for aerosol particles to become electrically charged is expected through radioactive decay, such as the loss of an electron in beta decay, and the ionization of gas in the intense radiation field. Based on Reference 7.2.49, aerosol removal rates of charged aerosols are greater than that of neutral particles. {{

}}2(a),(c)

4.3.8 Aerosol Plugging

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}}2(a),(c)

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. {{

}}2(a),(c)

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

{{

}}2(a),(c) All benchmark models satisfied the acceptance criteria. 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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}}2(a),(c)

{{

}}2(a),(c)

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

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}}2(a),(c)

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

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}}2(a),(c)

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

{{

}}2(a),(c)

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

4.3.10 Benchmarking to MAEROS

NMAEROS, a component of MELCOR, is the NuScale version of MAEROS (October, 1982). NMAEROS is equivalent to MAEROS (October, 1982) with input/output edit differences only. MAEROS is used to simulate the evolution of an aerosol based on the particle mass. Physical phenomena included in the calculations are coagulation, sedimentation, and diffusion. Thermophoresis was not modeled. 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. The aerosol sections, or bins, are defined by particle mass in MAEROS, as opposed to particle size as in STARNAUA. As a result, the numerical solution of the code, including time step sizes, is fundamentally different than that of STARNAUA. However, both codes utilize similar mechanics for coagulation and diffusion. As such, NMAEROS is appropriate for an independent code-to-code benchmark of STARNAUA with respect to the numerical solution of aerosol coagulation and diffusion.

For the benchmark, a hypothetical test roughly inspired by the LACE experiment was crafted. In order to compare the models, the hygroscopic, diffusiophoretic, and leak models in STARNAUA were turned off. Thermophoresis in both models was turned off. NMAEROS has a turbulent coagulation model, which should not have any significant impact for the particle size range under consideration (Reference 7.2.43, Figure 5-7). The only difference between Case 1 and 2 is the aerosol source rate. Case 1 is prescribed a CsOH source rate of 6.15 grams/sec while Case 2 provides a source rate of 0.615 grams/sec. The test geometry that the geometric inputs are derived from is shown in the following table. CsOH injection occurs from time zero to 1800 seconds in both cases.

Table 4-7. Test geometry

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}}2(a),(c)

{{

}}2(a),(c)

Figure 4-16. Benchmark Case 1 CsOH suspended concentration

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}}2(a),(c)

{{

}}2(a),(c)

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-NMAEROS 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, sedimentation, and diffusion.

4.3.11 Application

The description of the methodology to use STARNAUA for the CDE radiological consequence analysis 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 {{

}}2(a),(c)

{{

}}2(a),(c) For a given time step, the average release rate of radioactivity over a period of time can be evaluated by

$$R_s = A/D \qquad \text{Eq 4-14}$$

where R_s = release rate of radioactivity (*Ci/s*),

A = radioactivity (*Ci*),

D = total duration of release (*s)*.

Isotopic inventories source terms are provided in units of curies (Ci). For the purpose of modeling aerosol transport and removal, the mass is the physical property of concern. To convert from radioactivity (Ci) to mass (g), the relationship can be derived as

$$\because A\left(3.7 \times 10^{10}\,\frac{disintegrations}{s}/Ci\right) = \lambda N = \frac{\ln(2)}{T}\frac{m}{W_M}N_0 \qquad \text{Eq 4-15}$$

$$\therefore m = \frac{ATW_M}{N_0\ln(2)\,3.7 \times 10^{10}\,\frac{\frac{1}{s}}{Ci}} = 5.34 \times 10^{10}\,\frac{TW_M}{N_0}A \qquad \text{Eq 4-16}$$

where,

m = release aerosol mass (*g)*,

N = number of radioactive atoms,

λ = decay constant (s^{-1}) = *ln*(2)/T,

T = half-life of radionuclide (*s*),

$3.7 \times 10^{10}\,\frac{1}{s}/Ci$ = conversion factor from Ci to per second (*Bq*),

W_M = molecular (atomic) weight (*g/mol*), and

N_0= Avogadro's number = 6.022×10^{23} mol^{-1}

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}}2(a),(c)

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}}2(a),(c)

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}}2(a),(c)

For modeling aerosol, STARNAUA also requires the input of density and molecular weight for each aerosol species. Eight aerosol species are modeled; each corresponding to a chemical group. Therefore, each species is a mixture of various elements with all associated isotopes. For each species (a mixture), the effective molecular weight of an isotopic mixture can be weighted as

$$\bar{W}_M = MM \cdot \frac{\sum_{i=1}^{n} m_i}{\sum_{i=1}^{n} \frac{m_i}{W_{M,i}}} \qquad \text{Eq 4-22}$$

where *MM* = average mass multiplier of an isotopic mixture.

Similarly, the effective density of the species can be obtained by

$$\bar{\rho}_M = PF \cdot \frac{\sum_{i=1}^{n} m_i}{\sum_{i=1}^{n} v_i} = PF \cdot \frac{\sum_{i=1}^{n} m_i}{\sum_{i=1}^{n} \frac{m_i}{\rho_i}} \qquad \text{Eq 4-23}$$

where *PF* = packing factor.

It is noted that the apparent density used in the above equation is equal to the tabulated density multiplied by a packing factor. However, the density and molecular weight of an aerosol species is also dependent on the chemical form. In Table 13 of NUREG-6189 (Reference 7.2.49), possible chemical forms of aerosol species released into the reactor containment are listed with the mass multipliers of molecular weight. The mass multiplier for a chemical compound is defined as the molecular weight of the compound over the atomic weight of the element. For a polyatomic compound of *k* atoms, the mass multiplier of molecular weight is:

$$MM = \sum_{j=1}^{k} W_{M,j} / W_M \qquad \text{Eq 4-24}$$

where,

$W_{M,j}$ = molecular weight (atomic mass) of atom *j* in the compound (*g/mol*),

W_M = molecular (atomic) weight of the source nuclide.

This average mass multiplier is estimated for each chemical group in the aerosol modeling. The effective molecular weight for an aerosol species in Eq 4-22 can be obtained by modifying with the mass multiplier. The range (minimum and maximum) of molecular mass multipliers to account for the uncertain chemical form of radionuclides defined in Table 13 of NUREG/CR-6189 are reproduced in Table 4-8 with an additional column of the average value between the maximum and minimum. As the molybdenum group was not provided in NUREG/CR-6189, and at the time was included in the noble metals group, the value of the noble metals group is utilized for the molybdenum group.

Table 4-8. Radionuclide group molecular mass multipliers

Group	Min	Max	Average
Halogens	1.00	1.38	1.19
Alkali Metals	1.05	1.22	1.14
Tellurium Group	1.00	1.25	1.13
Alkaline Earths	1.11	1.67	1.39
Molybdenum	1.00	1.47	1.24
Noble Metals	1.00	1.47	1.24
Lanthanides	1.11	1.17	1.14
Cerium	1.17	1.22	1.20

Section II.G.5 of NUREG-6189 (Reference 7.2.49) and Reference 7.2.43 provide aerosol densities and molecular weights of principal interest. As a result, the physical properties of containment aerosols, the densities and molecular weights can be calculated for input into STARNAUA, from which the containment aerosol removal rate is determined.

4.4 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 CDE. 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 CDE, 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:

- **Boric Acid:** Boron is added to the RCS in the form of boric acid in order to control core reactivity. Although its concentration varies throughout the fuel cycle, {{ }}2(a),(c) Boric acid is a weak acid whose strength is dependent on temperature.
- **Lithium Hydroxide:** Lithium hydroxide is assumed to be added to the RCS in order to neutralize the acidic environment caused by boric acid. The amount of lithium hydroxide does not change from its initial value at a specific point in time when this analysis is performed. Lithium hydroxide is a strong base.
- **Hydriodic Acid:** Of the iodine released from the core as part of the CDE, 95 percent is assumed to be in the form of cesium iodide while the remaining 5 percent is assumed to either be hydriodic acid or organic iodine (Reference 7.2.52). Hydriodic acid is a strong acid.
- **Cesium Hydroxide:** Cesium released from the core as part of the CDE enters the coolant as either cesium iodide or cesium hydroxide. Some of the released cesium hydroxide may react with boron oxides and form cesium borates (Reference 7.2.52). Cesium hydroxide is a strong base.
- **Nitric Acid:** Nitric acid is generated by a radiolysis reaction in the liquid coolant (Reference 7.2.52). The irradiation of water at a water/air interface results in the generation of Nitric Acid. The amount of nitric acid in the coolant continually grows over the 30 day period. Nitric acid is a strong acid.
- **Hydrochloric Acid:** Hydrochloric acid is formed by a radiolysis reaction in power cable insulation that contains halogens (Reference 7.2.52). If halogenated cable insulation were used in the design that implements this methodology, then the amount of hydrochloric acid in the coolant would continually grow over the 30 day period. Hydrochloric acid is a strong acid.
- **Temperature:** The temperature of the coolant controls the neutral pH_T of the system. As temperature increases, the neutral pH_T decreases. At 100 degrees Celsius, the neutral pH_T of an aqueous system will be {{ }}2(a),(c)

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

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

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

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

4.4.1 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

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

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

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

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

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

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

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

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

The value of *K* for a specific dissociation reaction can be measured experimentally. Dissociation constant values for water, lithium hydroxide, and boric acid are taken from Reference 7.2.54.

Table 4-9. Dissociation constants of assumed acids and bases

Reaction Equation	Dissociation Constant, K
$H_2O \leftrightarrow H^+ + OH^-$	$\mathrm{Log}(K_{water}) = -14.9378 + 0.0424044 * T(°C) - 2.10252E\text{-}4 * T^2(°C)$ $+6.22026E\text{-}7 * T^3(°C) - 8.73826E\text{-}10 * T^4(°C)$
$LiOH \leftrightarrow Li^+ + OH^-$	$\mathrm{Log}(K_{LiOH}) = -0.7532 - 0.0048 * T(°C) + 6.746E\text{-}6 * T^2(°C)$
$B(OH)_3 + OH^- \leftrightarrow B(OH)_4^-$	$\mathrm{Log}(K_{B1}) = \frac{1573.21}{T(°K)} + 28.8397 + 0.012078 * T(°K) - 13.2258 * \log T(°K)$
$2\,B(OH)_3 + OH^- \leftrightarrow B_2(OH)_7^-$	$\mathrm{Log}(K_{B2}) = \frac{2756.1}{T(°K)} - 18.996 + 5.835 * \log T\,(°K)$
$3\,B(OH)_3 + OH^- \leftrightarrow B_3(OH)_{10}^-$	$\mathrm{Log}(K_{B3}) = \frac{3339.5}{T(°K)} - 8.084 + 1.497 * \log T(°K)$
$4\,B(OH)_3 + 2\,OH^- \leftrightarrow B_4(OH)_{14}^{-2}$	$\mathrm{Log}(K_{B4}) = \frac{12820}{T(°K)} - 134.56 + 42.105 * \log T(°K)$

4.4.2 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

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

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

4.4.3 Charge Balance Equation

The charge balance equation shows that the net charge of positive ions must equal the net charge of negative ions. In an aqueous solution consisting of a single acid and base, the charge balance equation is

$$[H^+] + [B^+] = [OH^-] + [A^-] \quad \text{Eq 4-35}$$

pH_T can be calculated by solving the charge balance equation for H^+. This calculation requires that equations are derived for B^+, OH^-, and A^- in terms of H^+. The value of OH^- can be found by rewriting Eq 4-30.

$$[\mathrm{OH^-}] = \frac{\mathrm{K_{H_2O}}}{[\mathrm{H^+}]} \qquad \text{Eq 4-36}$$

The value of A^- can be found by substituting *HA* from Eq 4-34 into Eq 4-32 and solving for A^-.

$$\mathrm{K_{HA}} = \frac{[\mathrm{H^+}] * [\mathrm{A^-}]}{[\mathrm{A}]_\mathrm{T} - [\mathrm{A^-}]} \qquad \text{Eq 4-37}$$

$$\mathrm{K_{HA}} * [\mathrm{A}]_\mathrm{T} = [\mathrm{A^-}] * [\mathrm{H^+}] + [\mathrm{A^-}] * \mathrm{K_{HA}} \qquad \text{Eq 4-38}$$

$$[\mathrm{A^-}] = \frac{\mathrm{K_{HA}} * [\mathrm{A}]_\mathrm{T}}{([\mathrm{H^+}] + \mathrm{K_{HA}})} \qquad \text{Eq 4-39}$$

Similarly, the value of *B+* can be found by substituting *BOH* from Eq 4-35 into Eq 4-33 and using Eq 4-37 to substitute for OH^-.

$$[\mathrm{B^+}] = \frac{\mathrm{K_{BOH}} * [\mathrm{B}]_\mathrm{T}}{\left(\mathrm{K_{H_2O}}/[\mathrm{H^+}] + \mathrm{K_{BOH}}\right)} \qquad \text{Eq 4-40}$$

With Eq 4-36, Eq 4-39, and Eq 4-40 defining OH^-, A^-, and B^+ written in terms of H^+, the charge balance equation (Eq 4-35) can be written as

$$[\mathrm{H^+}] + \frac{\mathrm{K_{BOH}} * [\mathrm{B}]_\mathrm{T}}{\left(\mathrm{K_{H_2O}}/[\mathrm{H^+}] + \mathrm{K_{BOH}}\right)} = \frac{\mathrm{K_{H_2O}}}{[\mathrm{H^+}]} + \frac{\mathrm{K_{HA}} * [\mathrm{A}]_\mathrm{T}}{([\mathrm{H^+}] + \mathrm{K_{HA}})} \qquad \text{Eq 4-41}$$

Eq 4-41 can be solved for H^+ using a variety of numerical techniques. Once H^+ is found, the pH_T of the aqueous system is calculated using Eq 4-25.

4.4.4 Final Charge Balance Equation

The methodology described in the previous sections is expanded to include any number of acids and bases. Incorporating the ionic concentrations from lithium hydroxide, cesium hydroxide, hydrochloric acid, nitric acid, hydriodic acid, and boric acid into the charge balance equation gives

$$[H^+] + [Cs^+] + [Li^+] = [OH^-] + [Cl^-] + [I^-] + [NO_3{}^-] + [B(OH)_4^-]$$
$$+[B_2(OH)_7^-] + [B_3(OH)_{10}^-] + 2 * [B_4(OH)_{14}^{-2}] \quad \text{Eq 4-42}$$

The $B_4(OH)_{14}^{-2}$ ion is multiplied by two to account for the negative two charge of the ion.

4.4.5 Concentration of Ionic Species

In order to solve the final charge balance equation (Eq 4-42), equations for the concentration of each ionic species in the system must be derived in terms of H^+. A discussion of these equations is presented in the following section.

4.4.5.1 Equations for Concentration

The equations describing the concentrations of Li^+, Cs^+, I^-, Cl^-, and NO_3^- are presented in Table 4-10. The concentration of Li^+ was derived using methodology from Section 4.4.3. Concentrations of Cs^+, I^-, Cl^-, and NO_3^- are based on assumptions for dissociation. Cesium hydroxide is assumed to undergo 95 percent dissociation at all temperatures per Reference 7.2.55 in which cesium hydroxide is a strong base, which typically undergoes nearly 100 percent dissociation. Assuming 95 percent dissociation conservatively weakens cesium hydroxide compared to other strong acids in the coolant over all temperature ranges. The acids hydriodic acid, hydrochloric acid, and nitric acid are considered strong acids at all temperature ranges, and thus undergo 100 percent dissociation. Assuming full dissociation ensures that the amount of dissociation these acids undergo is over-estimated, and therefore, conservative.

Table 4-10. Concentration equations of included chemical species

Chemical	Aqueous Reaction	Concentration in Terms of [H+]	Equation Number
Lithium Hydroxide	$LiOH \leftrightarrow Li^{+} + OH^{-}$	$[Li^{+}] = \frac{K_{LiOH} * [Li]_{Total}}{K_{water}/_{[H^{+}]} + K_{LiOH}}$	Eq 4-43
Cesium Hydroxide	$CsOH \leftrightarrow Cs^{+} + OH^{-}$	$[Cs^{+}] = 0.95 * \ [Cs]_{Total}$	Eq 4-44
Hydriodic Acid	$HI \leftrightarrow H^{+} + I^{-}$	$[\mathrm{I}^{-}] = 1.0 * [\mathrm{I}]_{Total}$	Eq 4-45
Hydrochloric Acid	$HCl \leftrightarrow H^{+} + Cl^{-}$	$[\mathrm{Cl}^{-}] = 1.0 * [Cl]_{Total}$	Eq 4-46
Nitric Acid	$HNO_3 \leftrightarrow H^{+} + NO_3^{-}$	$[NO_3^{-}] = 1.0 * [NO_3]_{Total}$	Eq 4-47

Boric acid is unique from other chemical species in this calculation in that it can undergo one of four separate aqueous reactions (Reference 7.2.54). The equations for the concentration of each ionic species of boron are given in Table 4-11. These equations include a $B(OH)_3$ term, the solution of which is derived in the following section.

Table 4-11. Concentration equations of boric acid ionic species

Boric Acid Ion	Aqueous Reaction	Concentration in Terms of [H+]	Equation Number
$B(OH)_4^-$	$B(OH)_3 + OH^- \leftrightarrow B(OH)_4^-$	$[B(OH)_4^-] = K_{B1} * [B(OH)_3] * K_{water}/[H^+]$	Eq 4-48
$B_2(OH)_7^-$	$2\,B(OH)_3 + OH^- \leftrightarrow B_2(OH)_7^-$	$[B_2(OH)_7^-] = K_{B2} * [B(OH)_3]^2 * K_{water}/[H^+]$	Eq 4-49
$B_3(OH)_{10}^-$	$3\,B(OH)_3 + OH^- \leftrightarrow B_3(OH)_{10}^-$	$[B_3(OH)_{10}^-] = K_{B3} * [B(OH)_3]^3 * K_{water}/[H^+]$	Eq 4-50
$B_4(OH)_{14}^{-2}$	$4\,B(OH)_3 + 2\,OH^- \leftrightarrow B_4(OH)_{14}^{-2}$	$[B_4(OH)_{14}^{-2}] = K_{B4} * [B(OH)_3]^4 * \left(K_{water}/[H^+]\right)^2$	Eq 4-51

4.4.5.2 [B(OH)3] Solution

The term $B(OH)_3$ found in Eq 4-48 through Eq 4-51 can be solved using the mass balance equation for boric acid.

$$[\mathrm{B}]_{\mathrm{Total}} = [B(OH)_3] + [B(OH)_4^-] + [B_2(OH)_7^-] + [B_3(OH)_{10}^-] + [B_4(OH)_{14}^{-2}] \quad \text{Eq 4-52}$$

Eq 4-48 through Eq 4-51 can be substituted into Eq 4-52 to give

$$\begin{aligned}[\mathrm{Boron}]_{\mathrm{Total}} = {} & [B(OH)_3] + \left\{K_{B1} * [B(OH)_3] * \left(\frac{K_{H_2O}}{[H^+]}\right)\right\} \\ & + \left\{K_{B2} * [B(OH)_3]^2 * \left(\frac{K_{H_2O}}{[H^+]}\right)\right\} \\ & + \left\{K_{B3} * [B(OH)_3]^3 * \left(\frac{K_{H_2O}}{[H^+]}\right)\right\} \\ & + \left\{K_{B4} * [B(OH)_3]^4 * \left(\frac{K_{H_2O}}{[H^+]}\right)^2\right\}\end{aligned} \quad \text{Eq 4-53}$$

A variety of root finding methods can be applied to Eq 4-53 at a constant value of H^+ in order to converge a solution for $B(OH)_3$. The solution to $B(OH)_3$ can then be applied to Eq 4-48 through Eq 4-51.

4.4.6 Iodine Re-evolution

An estimate for iodine re-evolution was made using Figure 3-1 of NUREG/CR-5950 (Reference 7.2.52), shown as Figure 4-18 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.

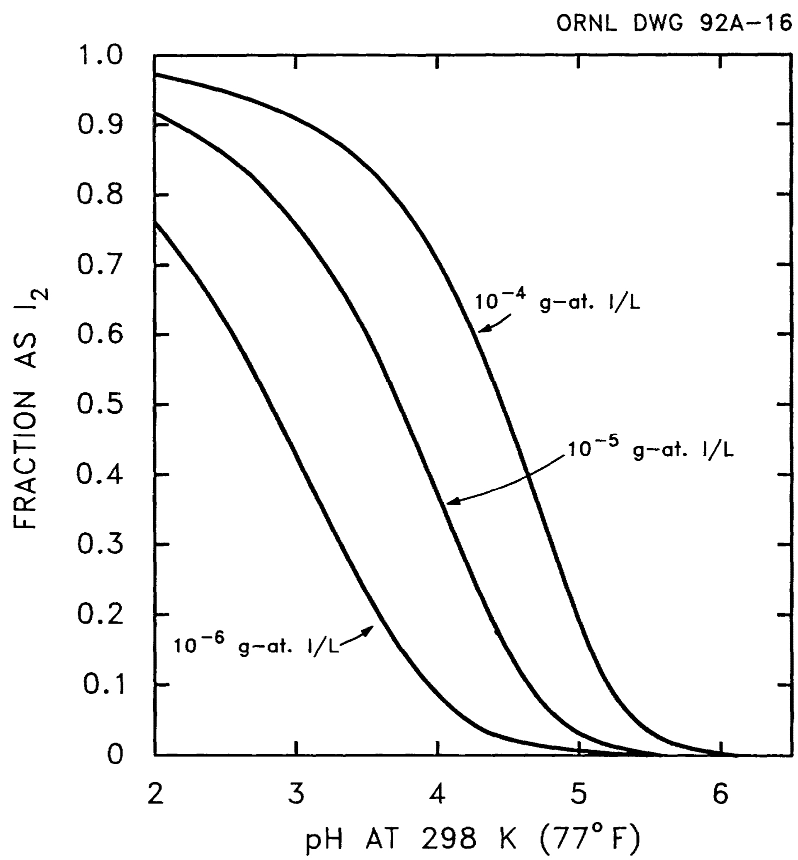

Figure 4-18. Iodine re-evolution versus pH (Reference 7.2.52)

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. Examples are provided in this section for offsite and onsite atmospheric dispersion factors, severe accident event selection, example severe accident analysis, containment aerosol removal, radiological consequence analyses, and post-accident pH_T. Some examples provided in Section 5 are based on a superseded preliminary version of the NuScale design. Since the purpose of these example results is illustrative and the changes in results would not be large enough to provide new insights into the application of the methodologies, the example results are not updated as methodology changes and revisions to this report occur. Differences in the methodologies originally utilized to create these example results and the methodologies stated in the current revision of this report are noted in the associated Section 5 subsections as appropriate.

5.1 Atmospheric Dispersion Factors

An applicant that implements this methodology is expected to use site-specific atmospheric dispersion factors calculated from qualified site-specific meteorological data obtained from a site specific RG 1.23 compliant meteorological monitoring program. In order to demonstrate the application of this methodology, this report assumed a three year data span for Sacramento, California from 1984 to 1986 in example calculations, described in Section 5.2 and Section 5.3. This representative site is assumed to occur on flat ground with nominal surface features (i.e., default surface roughness). An applicant who utilizes this methodology is expected to evaluate the applicability of the atmospheric dispersion modeling methodology for any significant site-specific geographical features. The example site information evaluated in this section is a representative example and is intended to illustrate how dispersion factors are calculated utilizing the methodology from Section 4.1 (with the exceptions of {{

}}2(a),(c)) as applied to a set of U.S. meteorological data.

In order to establish an appropriate site and associated meteorological dataset that could be used to develop atmospheric dispersion factors in a design certification, the methodology from Section 4.1 (with the exceptions of {{

}}2(a),(c)) is applied to a set of U.S. meteorological data from 241 sites across the U.S. from which a site representative of an 80-90th percentile U.S. site was selected; as recommended in the advanced light water reactor (ALWR) utility requirements document (URD), Reference 7.2.42 (which specifically recommended 80-90th percentile). The selected site meteorological data is then used in example calculations of offsite and control room atmospheric relative concentration values.

The EPA's Air Quality Modeling Group maintains the Support Center for Regulatory Atmospheric Modeling, a website that contains many tools related to atmospheric modeling (Reference 7.2.64). This website includes a database of hourly U.S. meteorological observations taken from the National Climatic Data Center (NCDC), which contains one or more years of surface meteorological data for various National Weather Service (NWS) stations across the U.S. The database also provides twice daily mixing height data for various NWS stations.

Both of the observations described above–surface and mixing height data–are required as input to PCRAMMET, a meteorological processor available on the EPA website. PCRAMMET processes the data and calculates stability class for each hour. The output file contains all of the hourly values required as input to ARCON96. The selected dataset contains 241 sites, each with three or more years of observations, and represents geographical diversity as shown in Figure 5-1.

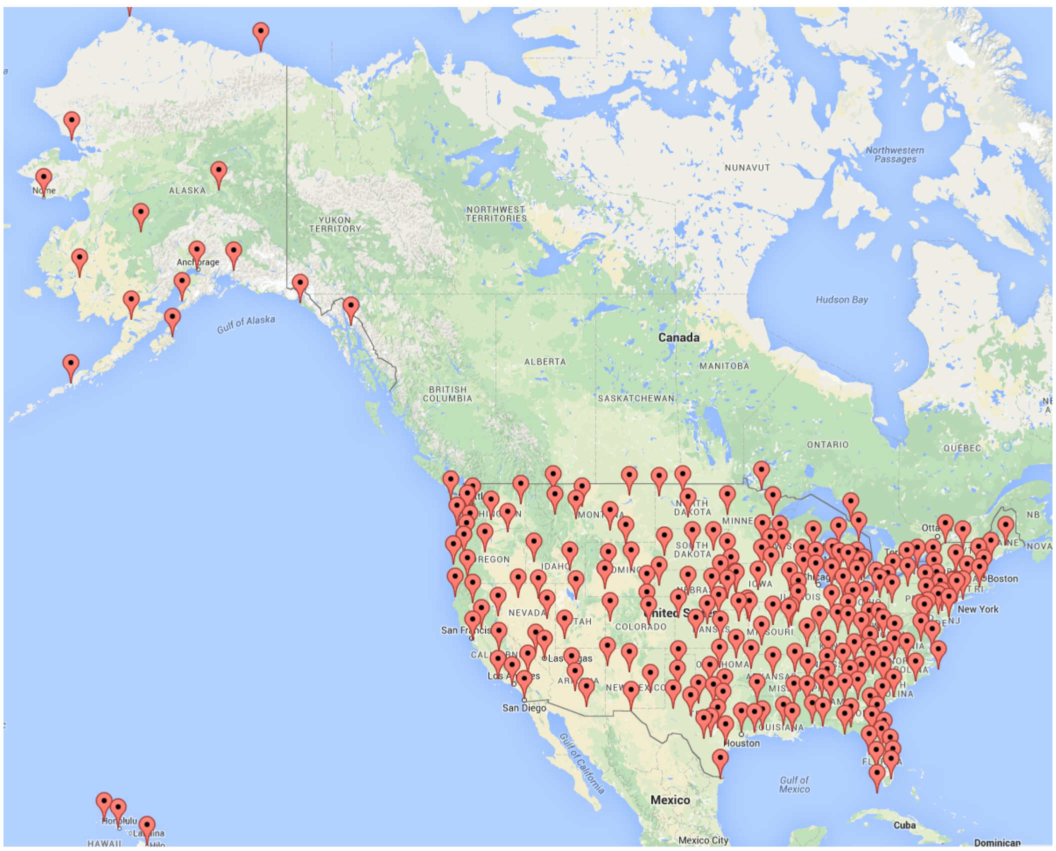

Figure 5-1. Map of each surface data site in the selected EPA dataset

The PCRAMMET method for calculating stability class uses the Pasquill-Gifford Turner method, which relies upon evaluation of solar insolation and cloud conditions to determine stability. This method is different from the delta-T method in RG 1.23. Both stability classes utilize seven stability classes (A thru G). Atmospheric dispersion factors are affected by the different methods, as each method is expected to produce different stability classes based on the same raw meteorological data (if the parameters for both methods existed in the data).

RG 1.23 recommends use of the delta-T method for determining Pasquill stability classes for licensing purposes. However, site-specific information is necessary in order to utilize the delta-T methodology. In the absence of this site-specific information, the Pasquill-Gifford Turner method is used in this example.

The Pasquill-Gifford Turner method utilized by PCRAMMET, while different from the delta-T method, is recommended by the EPA and only requires standard NWS observations, which are widely available.

Reference 7.2.63 performs a comparison of the two methods, with the delta-T method resulting in the highest frequency of limiting case stability (F, G) with a mean deviation between the two methods of 0.38. {{

}}2(a),(c)

All of the selected observations in the database were recorded between 1984 and 1992. There are more surface data sites than mixing height data sites; therefore, each surface data site is coupled with the nearest mixing height data site in order to produce the relevant inputs.

The selected database was originally recorded and published by the NCDC. Each site contains three or more years of data, and the database represents sufficient geographical diversity as illustrated in Figure 5-1. For these reasons, the 80-90th percentile of this database should sufficiently serve as meteorological data representative of an 80-90th percentile U.S. site.

5.1.1 Offsite Dispersion Factors

The postulated NuScale EAB and outer boundary LPZ distance is at the owner-controlled area boundary. As shown in Figure 5-2, there are several hypothetical locations that a release could occur. The selection of release location affects the distance between the release point and the EAB and LPZ, which is utilized to calculate the offsite dispersion factor. This distance could be recalculated on a case specific basis to provide realistic results, however, NuScale assumed the shortest distance of 400 feet in its example calculations to be conservative for all releases.

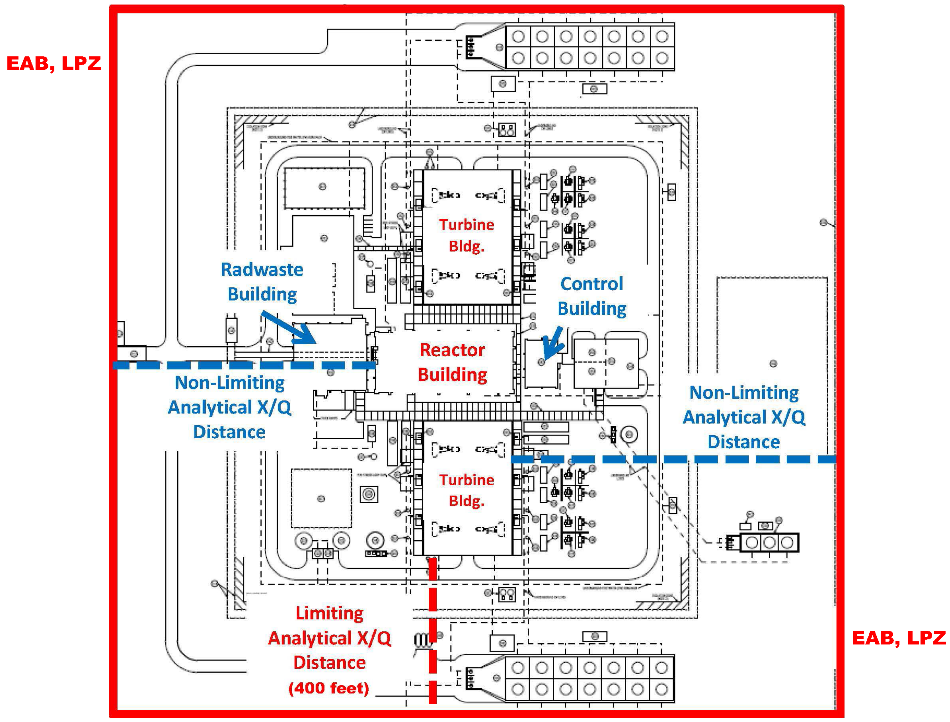

Figure 5-2. Markup of site layout with analytical offsite distances overlaid

The example analysis assumed a conservative cross sectional building area of 0.01 square meters, since smaller cross sectional building areas have been observed to produce larger relative concentrations. Note that ARCON96 has only one input for cross sectional building area and therefore this input accounts for the effect of all buildings between the source and receptor. All source geometries were assumed to be from a ground-level point source; no elevated, vent, heated, or diffuse sources were considered. The site terrain elevation differences were assumed to be zero.

ARCON96 was executed using data from each of the geographical sites in the selected meteorological database and executed 16 times for each site; once for each direction sector. The sectors are centered at ~22.5 degree intervals {{

}}2(a),(c) and each is 90 degrees in width, {{

}}2(a),(c). The maximum relative concentrations were selected for each site at each time period and distance. A set of 80th percentile relative concentrations and a set of 90th percentile relative concentrations were established by ordering the data from least to greatest and selecting the 80th and 90th percentile data points for each distance and each time period. Selection of an 80-90th percentile site is based on establishing a site whose relative concentrations typically fall between the 80th and 90th percentile relative concentration data sets.

Through this process the data span of 1984 to 1986 for the Sacramento, California site was chosen as the 80-90th percentile site. During this selection process importance was placed upon the 0 to 2 hour relative concentration as it typically dominates in dose calculations. Table 5-1 shows the selected site and its relative concentration for each time interval. Highlighted in grey in Table 5-1, the 8 to 24 hour interval of the selected site falls below the 80th percentile at some distances. This result is because the distribution of sites changes when downwind distance is changed. In other words, there is no site in the database that represents the 80-90th percentile at every distance and time interval. However, across all distances, this site is the most conservative and the best representation of the 80-90th percentile.

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

Downwind Distance (m)	0-2 hour	2-8 hour	8-24 hour	1-4 day	4-30 day
33	6.88E-03	5.65E-03	2.46E-03	2.47E-03	2.28E-03
66	1.86E-03	1.51E-03	6.72E-04	6.71E-04	6.13E-04
122	5.72E-04	4.84E-04	2.13E-04	2.15E-04	1.95E-04
201	2.28E-04	1.94E-04	8.46E-05	8.59E-05	7.83E-05
402	7.01E-05	5.89E-05	2.61E-05	2.55E-05	2.37E-05
805	3.44E-05	2.43E-05	1.09E-05	1.06E-05	9.51E-06
1609	2.28E-05	1.50E-05	5.95E-06	5.90E-06	5.30E-06

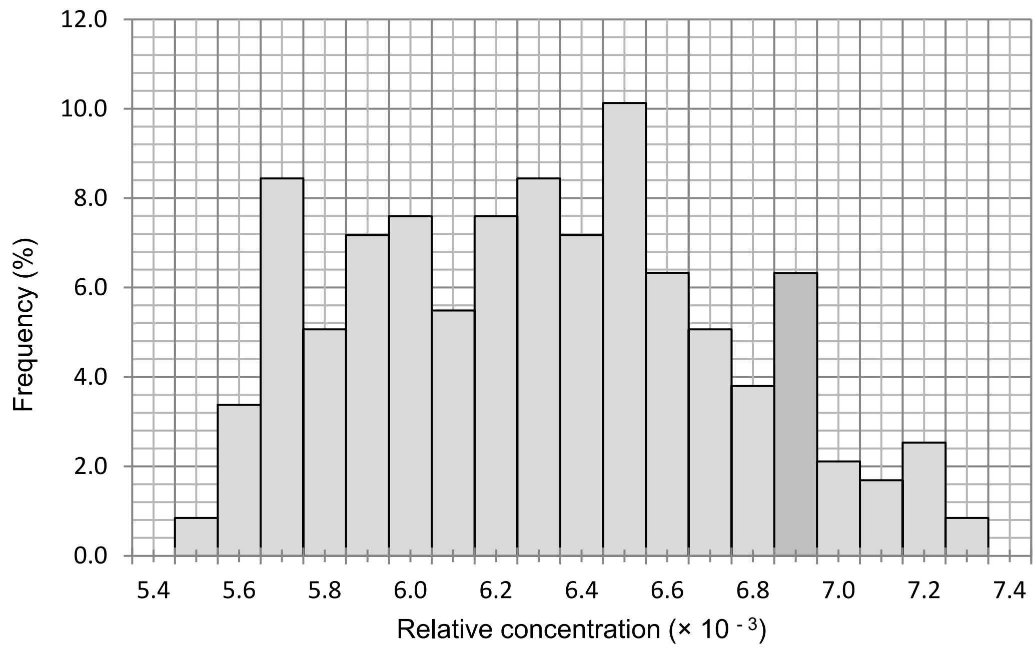

Figure 5-3. Histogram of calculation results at 33 meters

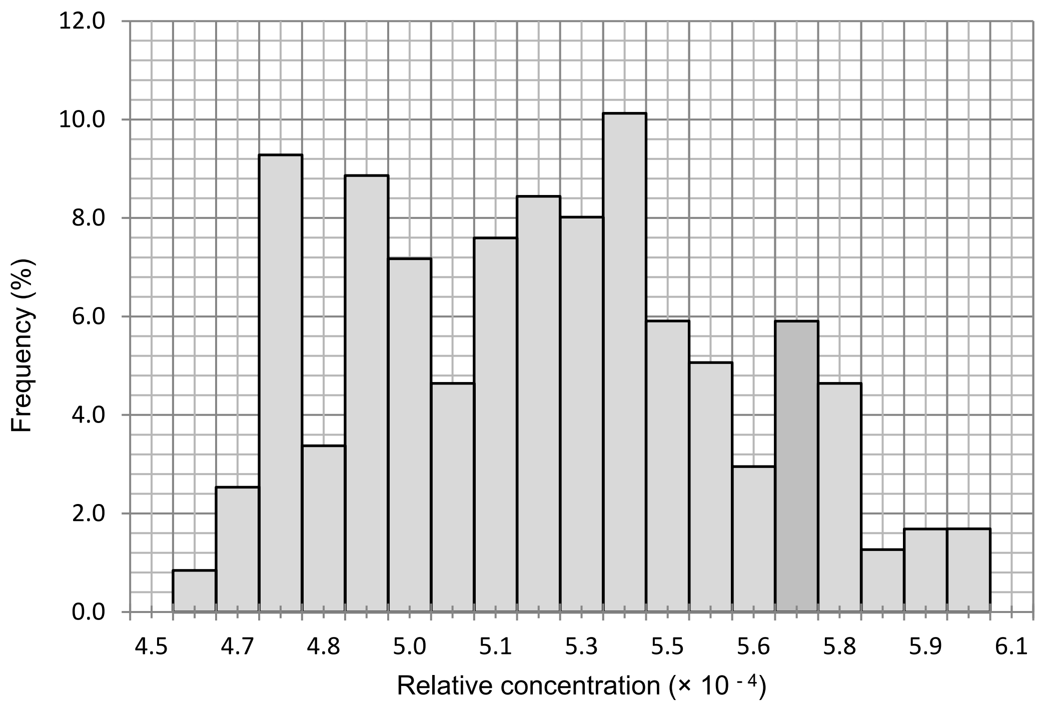

Figure 5-4. Histogram of calculation results at 122 meters

Table 5-2 presents the ratio of the selected relative concentration to the true 90th percentile relative concentration in the dataset. Though not all values represent the 80-90th percentile of the dataset, all of them are reasonably close in magnitude. There are five values below the 90th percentile of the dataset, but all of them are close to the 90th percentile in magnitude. Considering this relationship, and the fact that many of the selected concentrations are well above the 90th percentile, the selected site is justified for use as the 80-90th percentile of the dataset.

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

Downwind Distance (m)	0-2 hour	2-8 hour	8-24 hour	1-4 day	4-30 day
33	1.00	1.02	0.88	0.98	1.08
66	1.01	0.99	0.88	0.98	1.07
122	1.00	1.01	0.88	0.99	1.08
201	1.00	1.02	0.88	0.99	1.06
402	1.01	1.00	0.92	0.96	1.03
805	0.99	0.93	1.15	1.06	1.06
1609	0.90	0.96	1.25	1.08	1.13

Using the methodology specified in Section 4.1 of this report (with the exceptions of {{

}}2(a),(c), offsite atmospheric relative concentration values for the site located in Sacramento, California for the data span of three years (1984 to 1986) were calculated. These relative concentrations (shown in Table 5-4) are used in example dose calculations in this report. An applicant is expected to use site-specific atmospheric dispersion factors calculated from qualified site-specific meteorological data obtained from a site specific RG 1.23 compliant meteorological monitoring program. Use of NWS data is only for the purpose of illustrating the derivation of reasonable atmospheric relative concentration values for the example calculations in this report.

Table 5-3. Selected meteorological data

WBAN	Location	Number of Years	Span of Years
23232	Sacramento, California	3	1984-1986

The calculated offsite dispersion factors are presented in Table 3-2.

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

Distance (feet)	(meters)	0-2 hour (s/m^3)	2-8 hour (s/m^3)	8-24 hour (s/m^3)	1-4 day (s/m^3)	4-30 day (s/m^3)
400	121.9	5.72E-04	4.85E-04	2.14E-04	2.15E-04	1.95E-04

5.1.2 Control Room and Technical Support Center Dispersion Factors

Possible reactor or turbine building source locations, including doors, heating, ventilation, and air conditioning (HVAC) inlets and outlets, and penetrations, were examined for determining the limiting source locations. For the control room envelope and technical support center, personnel access doors and HVAC inlets were examined as possible receptor locations. In these example calculations, the control room ventilation air exhaust was not included as a control room receptor, because it was assumed that the control room emergency air will be continuously discharged through this location.

Utilizing the three dimensional coordinates provided by building drawings, the total and horizontal distances between source and receptor were calculated for each source-receptor combination. The total “taut-string” distance was considered as a vector length, therefore the standard equation for calculating vector lengths was utilized. The resultant control room atmospheric dispersion factors are presented in Table 5-5 for the limiting control room source-receptor distance.

Table 5-5. Example control room atmospheric dispersion factors

Distance (feet)	(meters)	0-2 hour (s/m^3)	2-8 hour (s/m^3)	8-24 hour (s/m^3)	1-4 day (s/m^3)	4-30 day (s/m^3)
111.78	34.1	6.27E-03	5.37E-03	2.31E-03	2.35E-03	2.13E-03

5.2 Design Basis Source Terms

Example dose results from the DBSTs described in Section 3.2 of this report are shown in Table 5-6. The acceptance criteria in Table 5-6 are taken from SRP Section 15.0.3. These example calculations utilized the dispersion factors associated with the limiting 80^{th}-90^{th} percentile site described in Section 5.1, and the general methodologies described in Section 3.3 (with the exception of the method utilized to calculate the iodine decontamination factor for the pool during a fuel handling accident), as applied to the example design assumed in these evaluations. For this example, the smallest margin between calculated and acceptance criteria dose is a factor of 7 smaller between the 5 Roentgen equivalent man (rem) control room dose acceptance criteria and the 0.72 rem calculated value.

A sensitivity study was performed for the SGTF and MSLB events assuming the liquid secondary coolant in the steam generator was at the primary coolant design basis limit concentration. This study resulted in an EAB dose increase of 1.4E-03 rem TEDE, as compared to the acceptance criteria of 2.5 rem or 25 rem, depending on the iodine spiking scenario assumed. Compared to the most limiting case from the applicable example calculations below, this result corresponds to a negligible 0.2 percent difference for the SGTF event with pre-incident iodine spiking.

Table 5-6. Example dose results for design-basis source terms

Event	Location	Acceptance Criteria (rem TEDE)	Dose (rem TEDE)
rod ejection accident (containment release)	EAB	6.3	0.011
	LPZ	6.3	0.144
	CR	5	0.131
rod ejection accident (primary system release)	EAB	6.3	0.001
	LPZ	6.3	0.001
	CR	5	0.004
fuel handling accident	EAB	6.3	0.362
	LPZ	6.3	0.362
	CR	5	0.313
main steam line break (pre-incident iodine spike)	EAB	25	0.004
	LPZ	25	0.019
	CR	5	0.023
main steam line break (coincident iodine spike)	EAB	2.5	0.0004
	LPZ	2.5	0.0014
	CR	5	0.0013
steam generator tube failure (pre-incident iodine spike)	EAB	25	0.637
	LPZ	25	0.663
	CR	5	0.720
steam generator tube failure (coincident iodine spike)	EAB	2.5	0.039
	LPZ	2.5	0.040
	CR	5	0.002
primary coolant line break	EAB	2.5	0.062
	LPZ	2.5	0.062
	CR	5	0.075
iodine spike DBST* (pre-incident iodine spike)	EAB	25	<0.01
	LPZ	25	<0.01
	CR	5	<0.01
iodine spike DBST* (coincident iodine spike)	EAB	25	<0.01
	LPZ	25	<0.01
	CR	5	<0.01

*Note: The iodine spike DBST is not an event, but rather a bounding source term associated with DBEs that result in primary coolant entering the containment.

5.3 Example Core Damage Source Term Selection Process

As described in Section 4.2 of this report, before establishing the CDST a spectrum of representative surrogate accident scenarios must be selected. With respect to the design evaluated in this report, the following is taken into consideration when establishing the proper spectrum of representative surrogate accident scenarios.

Based on severe accident analyses, all intact containment significant core damage scenarios involve the demand and failure of the emergency core cooling system (ECCS). Assessment of the ECCS has determined that incomplete actuation of the ECCS is both a more probable and more limiting failure mode than complete failure of all ECCS valves. However, ECCS failures consider the failure of all reactor recirculation valves (RRVs), all reactor vent valves (RVVs), or all ECCS valves.

In this example, the ECCS was demanded in one of two ways:

1) high liquid level in the containment

2) loss of DC power (LODC)

While there are numerous possible LOCA pathways into the containment, the most limiting assumed in this report is a break of the CVCS injection line. The inlet to this line is assumed to lie directly above the core in the riser and is assumed to be at a lower elevation than any other RPV penetration, other than the ECCS RRVs. As a result, a postulated break of the CVCS injection line leads to the most rapid loss of inventory of any LOCA evaluated and provides the most direct pathway for fission product release from the RPV.

A LODC is assumed to cause an ECCS demand because of the fail-safe design of the ECCS valves. In normal operation, the example assumes DC power energizes the solenoids that hold the valves in the closed position. Although a LODC is assumed to de-energize the solenoids and trip ECCS, the inadvertent actuation block[2] is assumed to prevent the valves from opening if the system is at or near operating pressure. The system is assumed to depressurize via either the decay heat removal system (DHRS) or cycling of the reactor safety valves before the ECCS valves open. The reactor safety valves are assumed to actuate only on high RPV pressure.

The example surrogate accident scenario spectrum selected to provide example CDST results are shown in Table 5-7. These selections are based on the aforementioned reasons. The example surrogate accident scenario cases shown in Table 5-7 cover a complete failure of ECCS and both types of incomplete ECCS actuation, and are assumed to initiate with either a LODC or a break of the CVCS injection line into the containment. No mitigating operator actions are considered and the availability of the DHRS is varied between the cases.

[2] Inadvertent actuation block is a design feature to prevent ECCS valves from opening during module operation, when high differential pressure exists across the valve.

Table 5-7. Spectrum of example surrogate accident scenario cases considered for creation of CDST

Case	Coolant Flow Path	ECCS
Case 1	CVCS injection line break	Complete failure
Case 2	CVCS injection line break	RVVs fail to open
Case 3	CVCS injection line break	RRVs fail to open
Case 4	Loss of DC power system	RVVs fail to open
Case 5	Loss of DC power system	RRVs fail to open

5.4 Example Severe Accident Analysis

Surrogate accident scenario case 2 is the most limiting of the five examples shown in Table 5-7 with respect to both time to core damage and release duration. Only the source term analysis associated with surrogate accident scenario case 2 is provided as an example in this report. Surrogate accident scenario case 2 is assumed to initiate with a break of the CVCS injection line at the top of the containment. Actuation of the ECCS is demanded by high liquid level in the containment. For this example sequence, the RVVs are assumed to fail while the RRVs open on demand. This incomplete ECCS actuation accelerates the loss of coolant to the containment until the core becomes uncovered and the fuel begins to heat up. However, the open RRVs result in reflooding of the core following the transport of noncondensable gas to the containment. The result is a limited release of fission products from the fuel and a short release duration.

Figure 5-5 through Figure 5-12 provide example results from the source term analysis associated with surrogate accident scenario case 2.

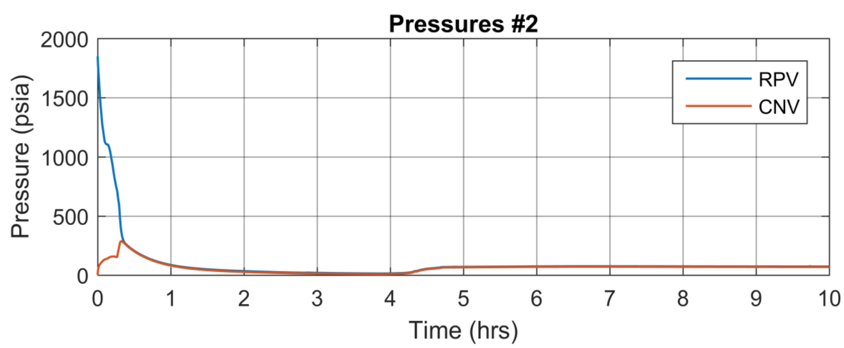

Figure 5-5. Example surrogate accident scenario case 2 short-term RPV and CNV pressures

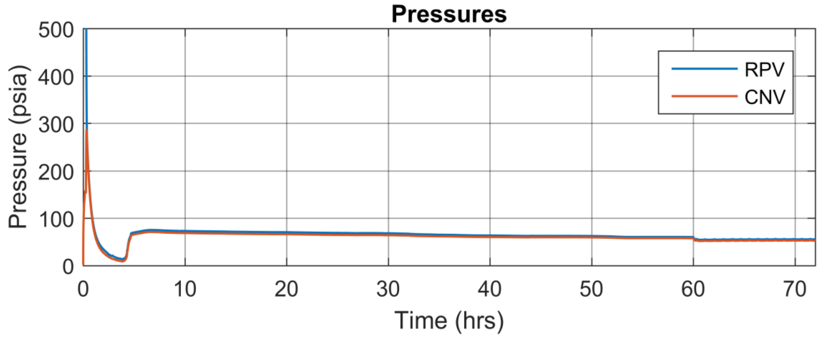

Figure 5-6. Example surrogate accident scenario case No. 2 long-term RPV and CNV pressures

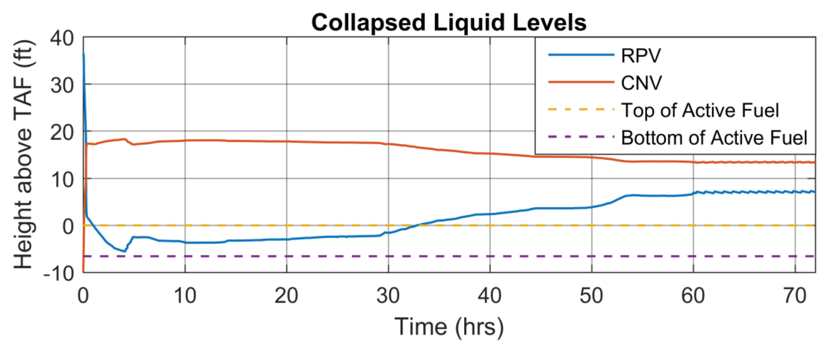

Figure 5-7. Example surrogate accident scenario case 2 RPV and CNV collapsed liquid levels

Figure 5-8. [Reserved]

Figure 5-9. [Reserved]

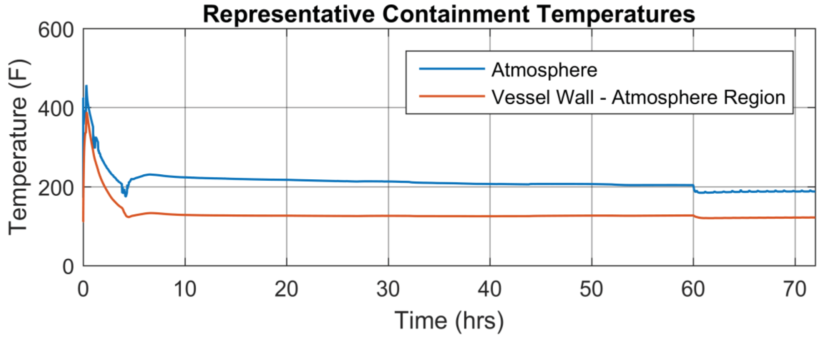

Figure 5-10. Example surrogate accident scenario case 2 representative containment temperatures

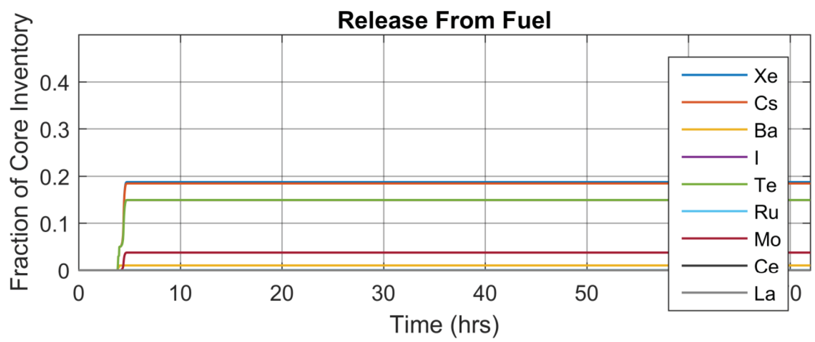

Figure 5-11. Example surrogate accident scenario case 2 release fractions from fuel

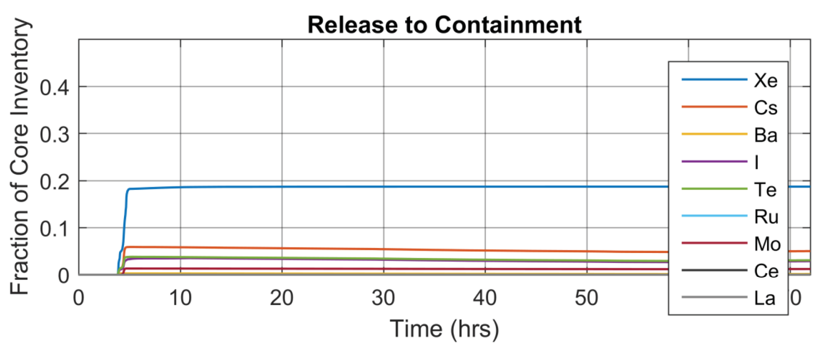

Figure 5-12. Example surrogate accident scenario case No. 2 release fractions into containment

5.5 Representative Severe Accident Results

Table 5-8 provides relevant fission product release information generated from MELCOR for the five example surrogate accident scenario cases. The fission product release information for each of the example surrogate accident scenario cases is compared to the values from RG 1.183 and the low burnup, low enriched uranium release fractions from SAND2011-0128.

For interpreting NuScale specific MELCOR results, the following definitions are provided:

- onset of gap release: assumed to occur at the start of clad ballooning as in SAND2011-0128
- duration of gap plus early in-vessel release: the first cladding failure to time when the release from the fuel ends

Table 5-8. Comparison of release timing and magnitudes of example surrogate accident scenario cases

Description		Case 1	Case 2	Case 3	Case 4	Case 5	Median Value	RG 1.183 Current	SAND 2011-0128
onset of gap release (hr)		17.6	3.8	8.1	6.2	21.3	8.1	30 sec	30 sec
duration of gap plus early in-vessel release (hr)		12.0	1.0	9.0	1.3	14.0	9.0	1.8	5.63
fraction of initial core inventory released into containment	noble gases	0.39	0.19	0.41	0.19	0.48	0.39	1	0.872
	halogens	0.21	3.5E-2	0.16	1.9E-2	0.14	0.14	0.4	0.307
	alkali metals	0.25	5.9E-2	0.22	3.1E-2	0.20	0.20	0.3	0.235
	alkaline earths	5.9E-3	2.8E-3	6.7E-3	2.4E-3	5.3E-3	5.3E-3	0.02	0.0054
	tellurium group	0.22	3.8E-2	0.16	2.3E-2	0.15	0.15	0.05	0.267
	molybdenum	6.4E-2	1.3E-2	5.3E-2	5.8E-3	4.9E-2	4.9E-2	0.0025	0.1
	noble metals	1.2E-03	1.2E-4	1.5E-3	4.9E-5	7.9E-4	7.9E-4	0.0025	0.006
	lanthanides	3.3E-8	2.6E-9	3.1E-8	1.1E-9	2.1E-8	2.1E-8	0.0002	1.1E-07
	cerium group	3.3E-8	2.6E-9	3.1E-8	1.1E-9	2.1E-8	2.1E-8	0.0005	1.1E-07

As observed from Table 5-8, the nine example median release fractions of the radionuclide groups are bounded by those in SAND2011-0128. These NuScale design-specific median release fractions are used for the CDE radiological consequence example calculation in this report.

This onset time is delayed relative to RG 1.183 onset time, with a range of 3.8 to 21.3 hours. Similarly, the duration of release is similar to or much greater than RG 1.183 with calculated values of 1.0 to 14.0 hours.

5.6 Example Containment Aerosol Transport and Removal

Example containment aerosol transport and removal calculations, including sensitivity cases, are provided in this section. These example calculations utilize methodology described in Section 4.3 of this report, as applied to the example design. The injection rates, aerosol properties, and aerosol removal rates and concentrations were calculated using the previously discussed MELCOR and STARNAUA computer codes. Example surrogate accident scenario cases from Section 5.5 are utilized, with the release timing (onset and duration) and release fractions applied from Table 5-8. Aerosol parameters are taken from Table 4-5 and biased to a conservative direction as suggested by example Table 5-18. All other aerosol input parameters utilize the nominal or default values, which must be shown to be conservative, as appropriate, with respect to the parameter-specific bias direction in design calculations. The accident scenarios of concern are intact containment core damage with the median release fractions along with surrogate accident scenario case No. 2 used as the baseline case for aerosol study. Surrogate accident scenario case 5 is chosen for analysis in a sensitivity case because it is the most probable of the five surrogate accident scenario cases considered. Results of the baseline and sensitivity case are summarized in Table 5-9.

Table 5-9. Example accidents for aerosol simulation

Case Name	Onset of Release (hr)	Release Duration (hr)	Release Fractions	Pressure / Temperature
Baseline	3.8	1.0	Median	Case 2
Sensitivity	21.3	14.0	Case 5	Case 5

Figure 5-13 is a plot of containment aerosol concentration and removal as a function of time for the baseline case as calculated by STARNAUA. The end of aerosol release injection occurs at 4.8 hours, 3.8 hours after onset of event and 1 hour of release duration (3.8+1=4.8). Between 3.8 hours and 4.8 hours the linear injection of aerosol into containment results in growth of the containment concentration at a relatively constant rate. Removal is occurring, but is being outpaced by concentration addition during this injection period. At 4.8 hours with the end of release, a period of stable removal of aerosol through sedimentation and other mechanisms begins and results in an exponential decay of the concentration as the average size of the aerosol molecules increases exponentially, as displayed in Figure 5-14. While the relative removal rate remains stable, less and less aerosol is available to remove, and the aerosol concentration continues to decrease exponentially.

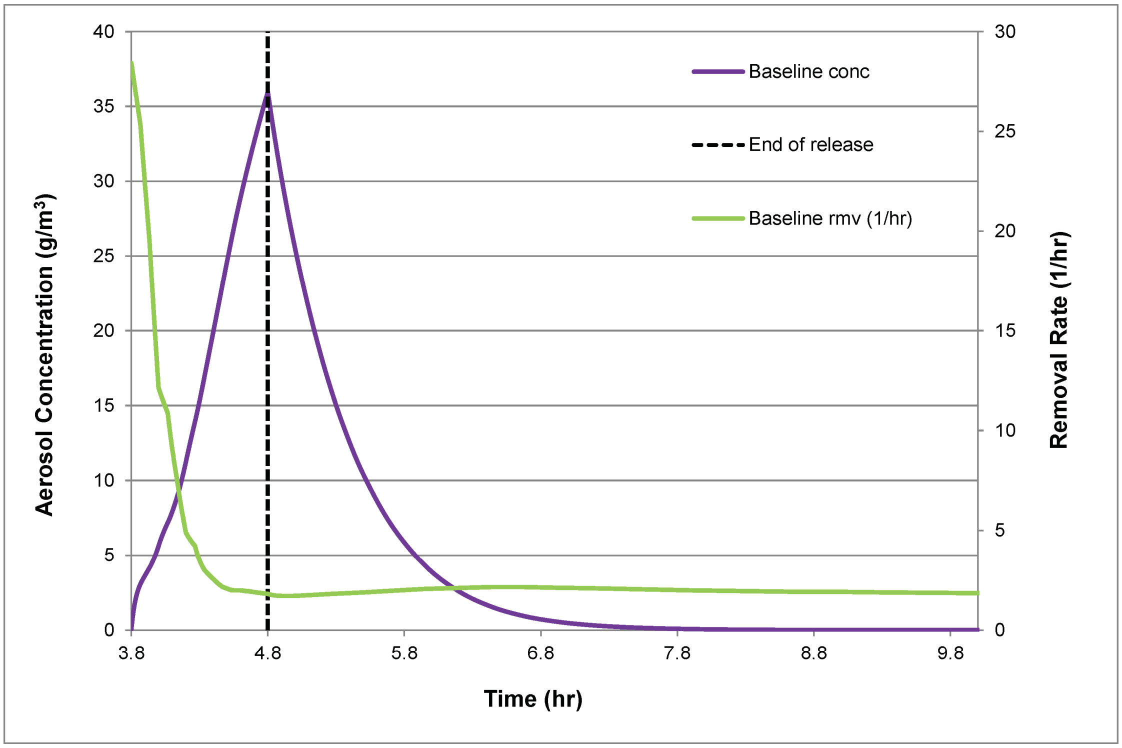

Figure 5-13. Baseline case aerosol concentration and removal

Figure 5-14 is a plot of the aerodynamic mass-median diameter of the aerosol molecules as a function of time. As expected, the growth is approximately logarithmic, with deviations from this trend during the injection period. During this time, aerosols already in the containment vessel likely are experiencing rapid growth in size, while the aerosols continuously being injected from the reactor vessel are at their original, and smaller, size.

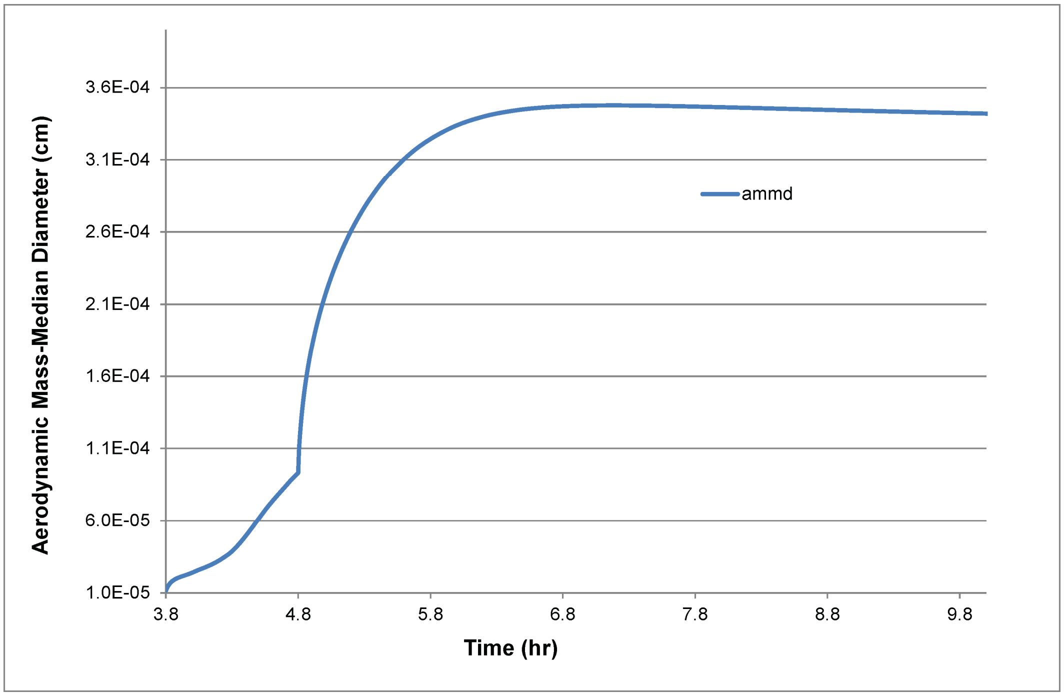

Figure 5-14. Baseline case aerosol aerodynamic mass-median diameter

Table 5-10 is a summary of example key concentration and removal rate parameters for all cases as calculated by STARNAUA. An unweighted average removal value is calculated for general information purposes.

Table 5-10. Summary of key parameters from all cases

Parameter	Average Removal	Maximum Concentration	Time of Max. Concentration
Units	(hr^{-1})	(g/m^3)	(hr)
Baseline	7.6	36.0	4.8
Sensitivity	1.3	19.0	29.8

Figure 5-15 and Figure 5-16 present plots of aerosol concentration and removal rate for all example cases, as calculated by STARNAUA.

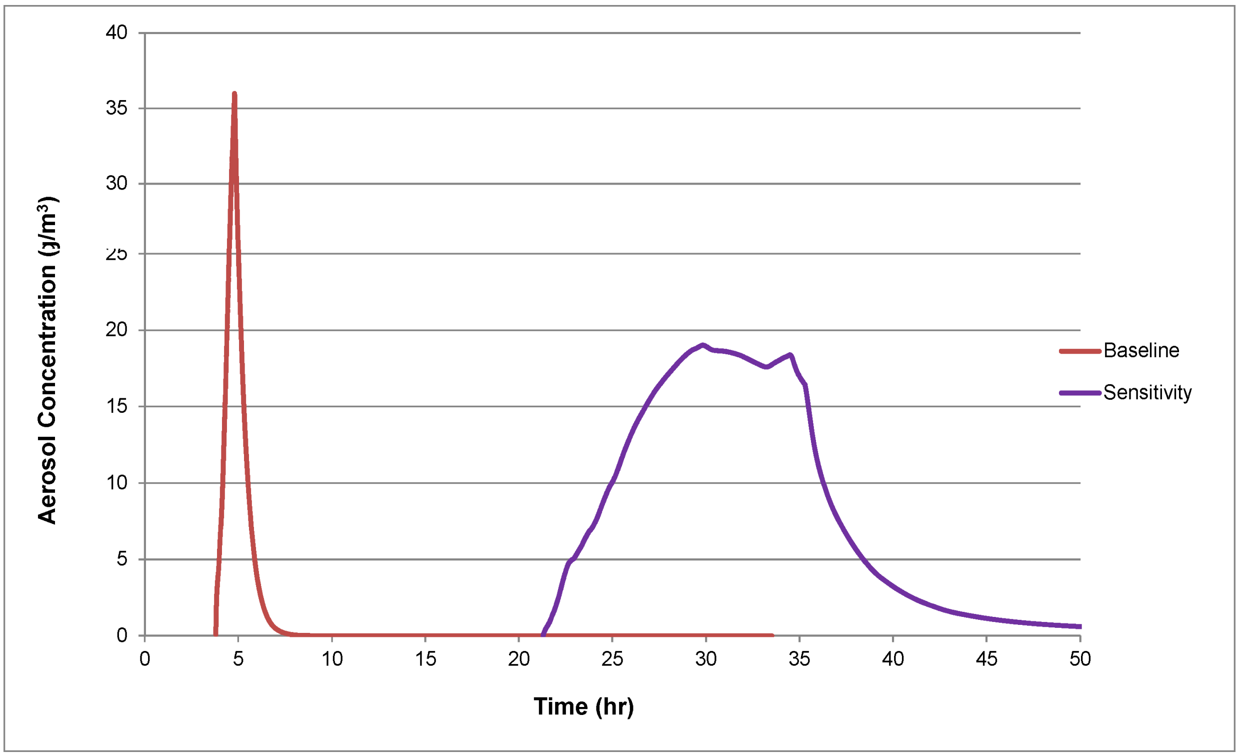

Figure 5-15. Comparison of aerosol concentration for all example cases versus time

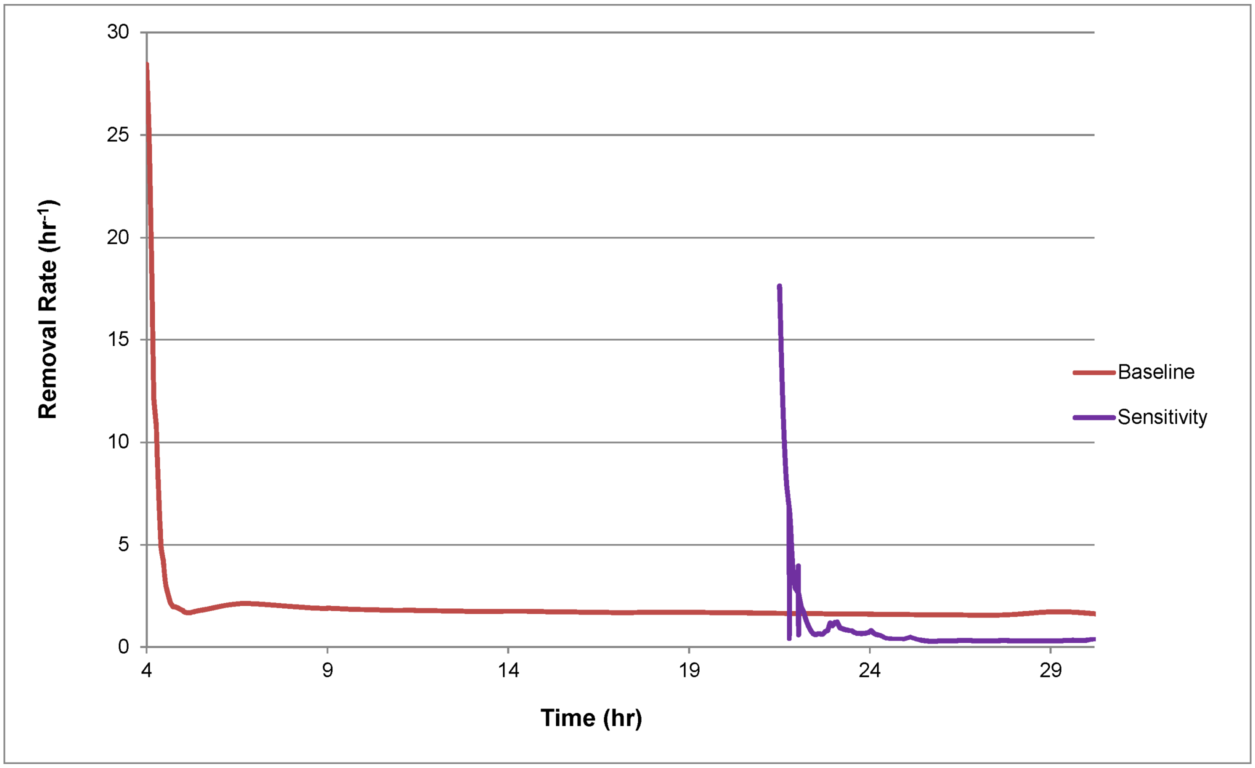

Figure 5-16. Comparison of aerosol removal rate for example cases versus time

An exponential decay of the postulated accident aerosol concentration through natural deposition removal mechanisms results in significant decontamination of airborne fission products in the containment vessel. Therefore, the example radiological consequences of the CDE are reduced as a result of the naturally occurring aerosol removal.

Table 5-11. Summary of example aerosol removal results

Parameter	Units	Value
Peak Concentration	g/cm^3	36.0
Average Removal Rate	hr^{-1}	7.6

5.7 Example Core Damage Event Radiological Consequences

With the example containment aerosol transport and removal calculations, including sensitivity case, the resulting control room and offsite radiological consequences are estimated in this section. These example calculations utilized dispersion factors based on meteorological data representative of an 80^{th}-90^{th} percentile site and the general methodologies described in Section 3.3 and Section 4.3, as applied to the example design. The same case matrix defined in Table 5-9 is utilized, with the example radiological consequences summarized in Table 5-12.

Table 5-12. Summary of example RADTRAD case results

Case Name	Release Fractions	Removal Rate	Release Onset (hr)	Release Duration (hr)	Dose (rem TEDE)		
					CR	EAB	LPZ
Acceptance Criteria					**<5**	**<25**	**<25**
Baseline	Median Value	Baseline	3.8	1.0	2.14	0.63	1.37
Sensitivity	Case 5	Sensitivity	21.3	14.0	1.78	0.20	1.28

In all of the example cases, the acceptance criteria were satisfied.

5.8 Core Damage Source Term Sensitivity Analysis

5.8.1 General Sensitivity Analysis Methodology

General sensitivity analysis methodologies are used throughout this section and are provided herein for reference. The sensitivity analyses are used in the accident radiological methodology to determine the appropriate biasing direction for an input, in order to obtain a conservative solution and to determine relative importance of the inputs. Another use of this sensitivity analysis is to check for the possibility of a non-linear system with corresponding feedback effects. Non-linear systems are checked for because parameter importance and bias directions are not consistent in such systems. Thus, in the case of a non-linear system, conditional bias directions have to be established in order to ensure a robust deterministic methodology. A review of the input parameters ranked in importance by linear, rank, and quadratic regression indicates clear consistency for each case individually, but also between different cases. This observation is a strong indication that the analysis of the NuScale design, as performed with a deterministic methodology, results in the modeled system behaving largely as a linear system (i.e., consistent importance and bias directions).

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.

5.8.2 Application to Core Damage Event

Example calculations shown throughout Section 5.8 are based on the aerosol modeling methodology shown in Revisions 0-2 of this topical report and have not been updated to be consistent with the aerosol methodology in the current revision of this report. General sensitivity analysis methodologies are applied to the CDE radiological consequence 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 CDE radiological consequence calculation, which is the dose.

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. Exclusion area boundary dose is not included as it focuses only on the limiting two hour window. As the LPZ is assumed to be at the same distance as the EAB for this methodology, the cumulative LPZ dose is considered representative of EAB results. This consideration is made because the 0 to 2 hour sliding window EAB value is a single number that could occur at any point during the 30 day period, as opposed to the transient LPZ dose that continues to increase throughout the event, albeit at different rates during the transient.

Table 5-13 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 5-13. 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. Non-linear systems are checked for because 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 5-14 and Table 5-15 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 5-14. Key control room dose input rankings and bias directions

Description	Conservative Bias	Rank	PRCC	Quadratic Incremental R^2
CR in-leakage	Larger	1	0.98	0.879
Release duration	Smaller	2	-0.56	0.038
CR volume	Smaller	3	-0.46	0.021
Release fractions	Larger	4	0.43	0.020
Isotopic activity	Larger	5	0.40	0.013
CNV leakage	Larger	6	0.40	0.011
CR *X/Q*	Larger	7	0.37	0.012

Table 5-15. Key low population zone dose input rankings and bias directions

Description	Conservative Bias	Rank	PRCC	Quadratic Incremental R^2
Release duration	Smaller	1	-0.90	0.522
CNV leakage	Larger	2	0.71	0.117
LPZ *X/Q*	Larger	3	0.70	0.121
Release fractions	Larger	4	0.69	0.121
Isotopic activity	Larger	5	0.69	0.106

Figure 5-17 and Figure 5-18 present example control room and LPZ dose results assuming constant aerosol inputs in trace plot format. These plots show the transient progression for each individual trial shown in gray (only the first 200 trials are plotted for readability). The black dashed lines in the figures indicate the minimum and maximum values at any time during the transient. The blue solid line is the average value across all trials. The red dots show the absolute peak value found during each trial (only the first 200 are plotted for readability).

Note that a significant inflection point in the example LPZ dose results occurs at eight hours. This occurrence is due to the change in atmospheric dispersion coefficients assumed in the period of 2 to 8 hours and 8 to 24 hours. There is no inflection point at 2 hours because the 0 to 2 hour and 2 to 8 hour factors vary by approximately 15 percent, within the range of the perturbation of these values and other parameters. However, the difference between the 2 to 8 hour and 8 to 24 hour value is significant (approximately 60 percent), and thus observable. This same inflection point is not observed in the example control room dose, even though there is a similar change in the dispersion factors, because the control room dispersion factors are of a much lower relative importance.

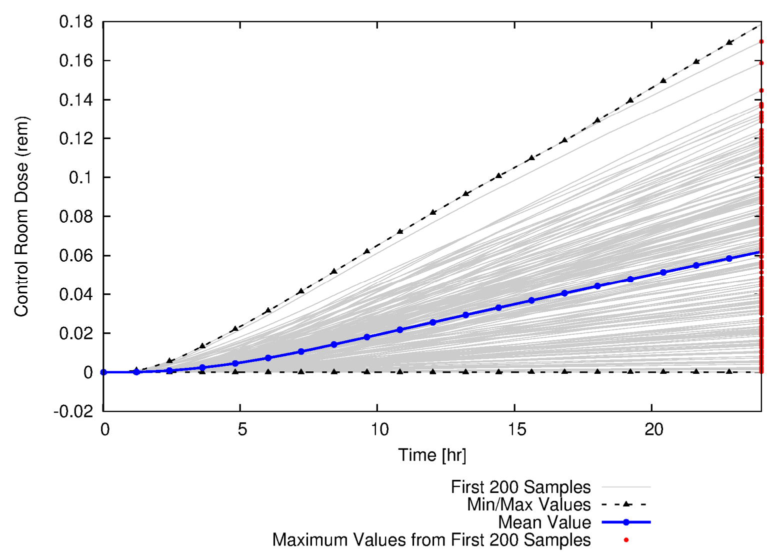

Figure 5-17. Example control room dose trace plot (constant aerosol)

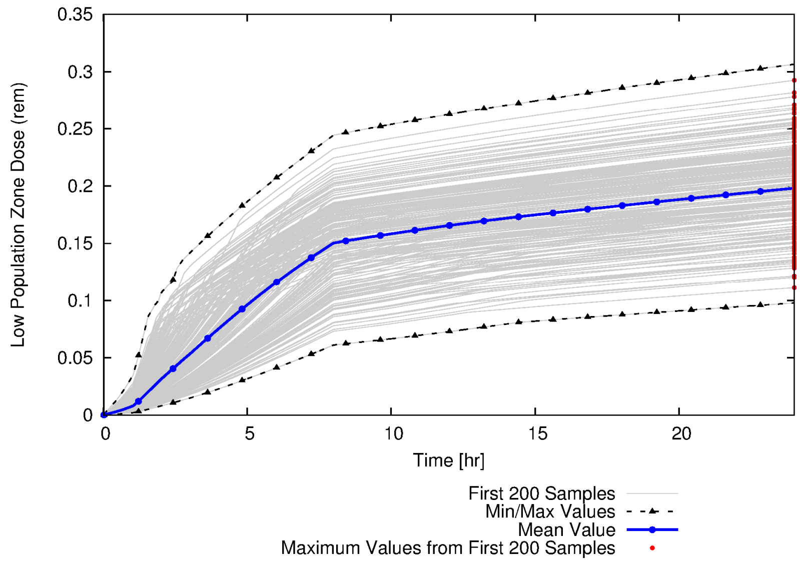

Figure 5-18. Example low population zone dose trace plot (constant aerosol)

Figure 5-19 and Figure 5-20 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 5-14 and Table 5-15 in a visual format.

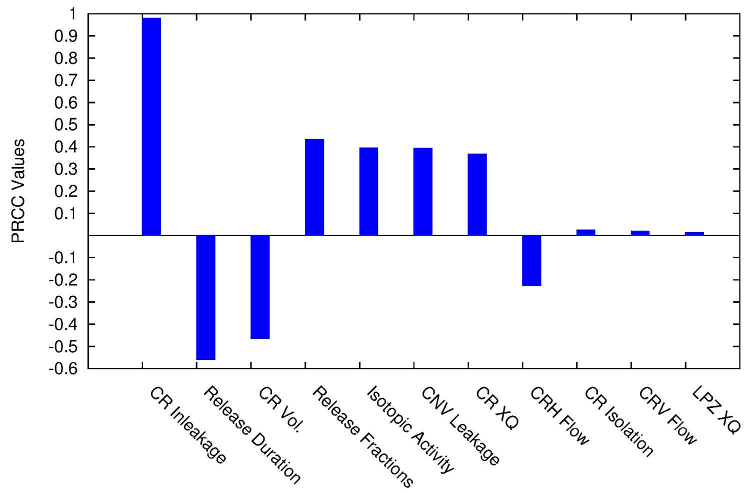

Figure 5-19. Example control room dose sensitivity rankings (constant aerosol)

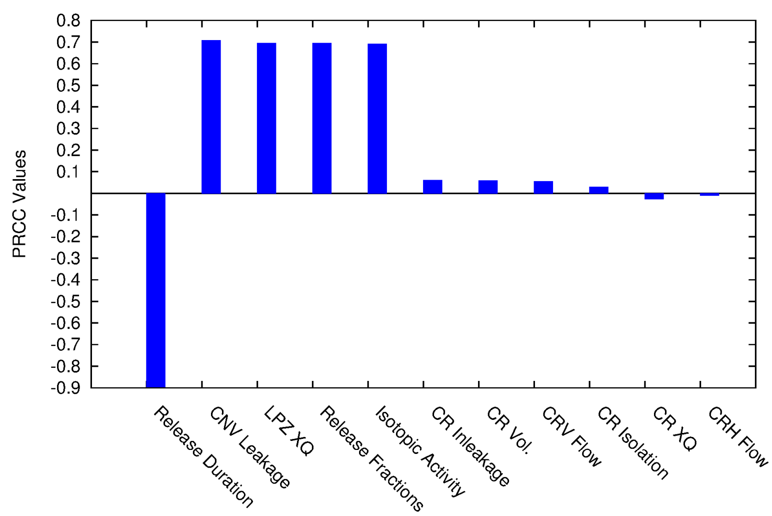

Figure 5-20. Example low population zone dose sensitivity rankings (constant aerosol)

Aerosol-specific inputs are added to the sensitivity analysis, with results presented in Table 5-16, Table 5-17, Figure 5-21, and Figure 5-22. 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 5-16. Key aerosol inputs for low population zone dose rankings and bias directions

Description	Conservative Bias	Rank	PRCC	Quadratic Incremental R^2
Release duration	Smaller	1	-0.77	0.117
Packing fraction	Smaller	2	-0.76	0.246
Mobility shape	Larger	3	0.68	0.102
Geometric standard deviation	Smaller	4	-0.63	0.100
Max bin radius	Smaller	5	-0.50	0.060
Inert ratio	Smaller	10	-0.37	0.026
Mean radius	Smaller	11	-0.36	0.038
CNV volume	Larger	12	0.32	<0.02
Sedimentation area	Smaller	13	-0.29	<0.02

Table 5-17. Key aerosol concentration input rankings and bias directions

Description	Conservative Bias	Rank	PRCC	Quadratic Incremental R^2
Release duration	Smaller	1	-0.90	0.470
Packing fraction	Smaller	2	-0.76	0.159
Mobility shape	Larger	3	0.71	0.079
Geometric standard deviation	Smaller	4	-0.66	0.078
Max bin radius	Smaller	5	-0.54	0.046
Mean radius	Smaller	6	-0.40	0.019
Sedimentation area	Smaller	7	-0.33	0.012
Inert ratio	Smaller	9	0.12	<0.02

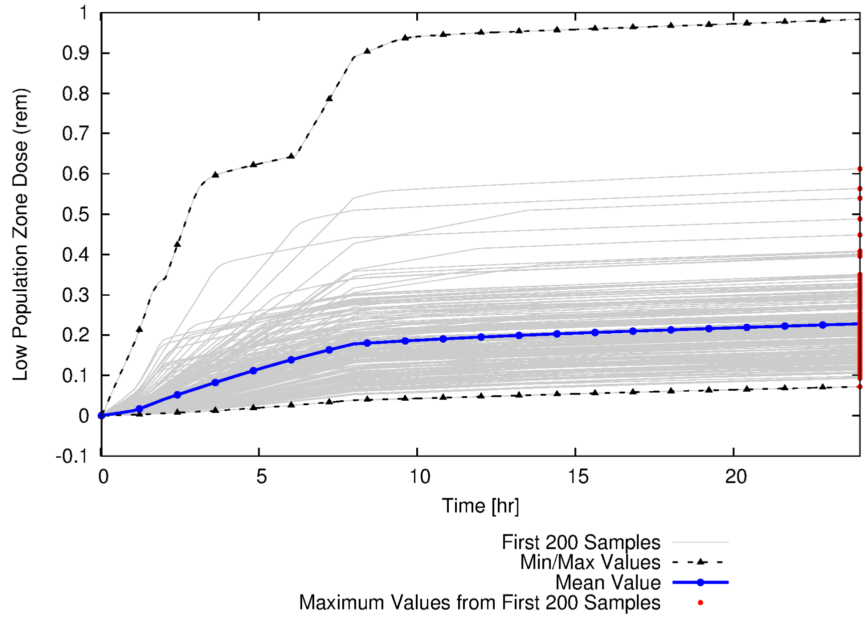

Figure 5-21. Low population zone dose trace plot

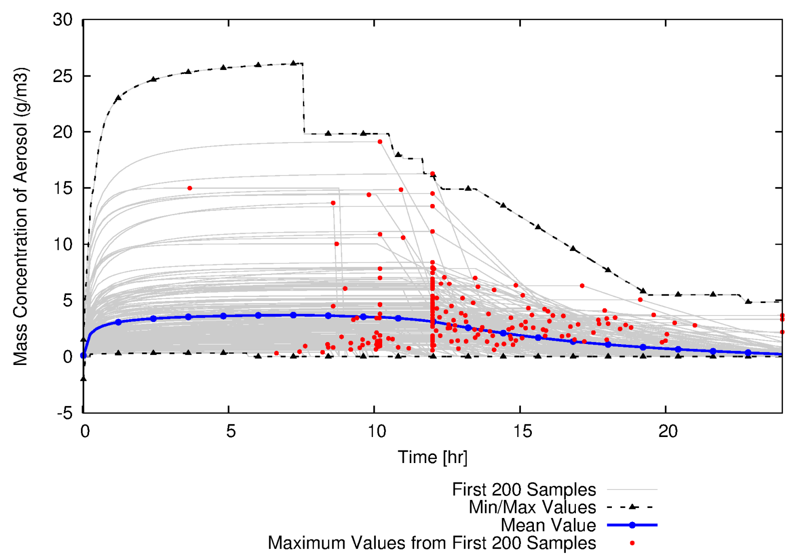

Figure 5-22. Aerosol concentration trace plot

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}}2(a),(c) Table 5-18 provides the sensitivity analysis established bias directions for parameters to maximize dose and minimize aerosol removal.

Table 5-18. Direction of bias to maximize dose and minimize aerosol removal

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5.8.3 Sensitivity Analysis Conclusions

Results from the example input sensitivity analysis performed for the CDE radiological consequence analysis 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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5.9 Post-Accident pH_T

For illustrative purposes, the methodology of Section 4.4 was implemented to a scenario representative of a surrogate accident scenario. An example baseline and sensitivity calculation is provided in this section for context. An example baseline calculation provides a minimum pH_T value of 6.7 inside containment, which is the assumed initial value. Note that at full operating conditions, the neutral pH_T of the coolant is assumed to be 5.6. An estimate for iodine re-evolution can be made using Figure 4-18. According to this figure, less than 1 percent of aqueous iodine is assumed to be converted into molecular I_2 at a pH_T of 6.0.

An example temperature sensitivity analysis demonstrates the importance of neutral pH_T when determining whether the environment in containment is acidic or basic. As the temperature in containment increases, the neutral pH_T of the coolant decreases while the actual value of pH_T has remained relatively constant. At temperatures above 100°C, the environment inside containment will be basic.

The results, calculated with the "pHT" computer code described in Section 3.1.7, of the example cases are summarized in Table 5-19. The example temperature sensitivity data is provided in Table 5-20 and includes how the neutral pH_T of the coolant changes in acidity.

Table 5-19. Summary of example pH_T results for calculations performed at 25 degrees C

Case Description	Initial pH_T	Maximum pH_T	Minimum pH_T
Base Line	6.7	7.1	6.7
1000 ppm Boron	6.9	7.4	6.9
500 ppm Boron	7.1	7.9	7.1
200 ppm Boron	7.3	8.3	7.3
50% of CsOH	6.7	6.9	6.7
150% of CsOH	6.7	7.2	6.7
No Nitric or Hydrochloric Acid	6.7	7.1	6.7
Only Hydrochloric Acid	6.7	7.1	6.7
Both Nitric and Hydrochloric Acids	6.7	7.1	6.7
45,000 kg Liquid Coolant	6.6	7.0	6.6
30,000 kg Liquid Coolant	6.3	6.7	6.3

Table 5-20. Summary of example results for baseline calculation with increasing temperatures

Case Description	Initial pH_T	Maximum pH_T	Minimum pH_T	Neutral pH_T	Acid/Base at Minimum pH_T
25° C	6.7	7.1	6.7	7.0	Acid
100° C	6.7	7.1	6.7	6.1	Base
200° C	6.9	7.3	6.9	5.6	Base

The example baseline calculation uses conservative assumptions and inputs, resulting in coolant pH_T plotted over 30 days (see Figure 5-23), with a final pH_T of 7.1. Example conservative inputs for the baseline case are a coolant temperature of 25 degrees C, coolant mass of 53,400 kg, boron concentration of 1500 ppm, lithium concentration of 9.14 ppm, maximum iodine and cesium mass, and design basis containment doses. The boron concentration is assumed to vary between almost 1300 ppm at the beginning of an operating cycle to 5 ppm at the end of the cycle.

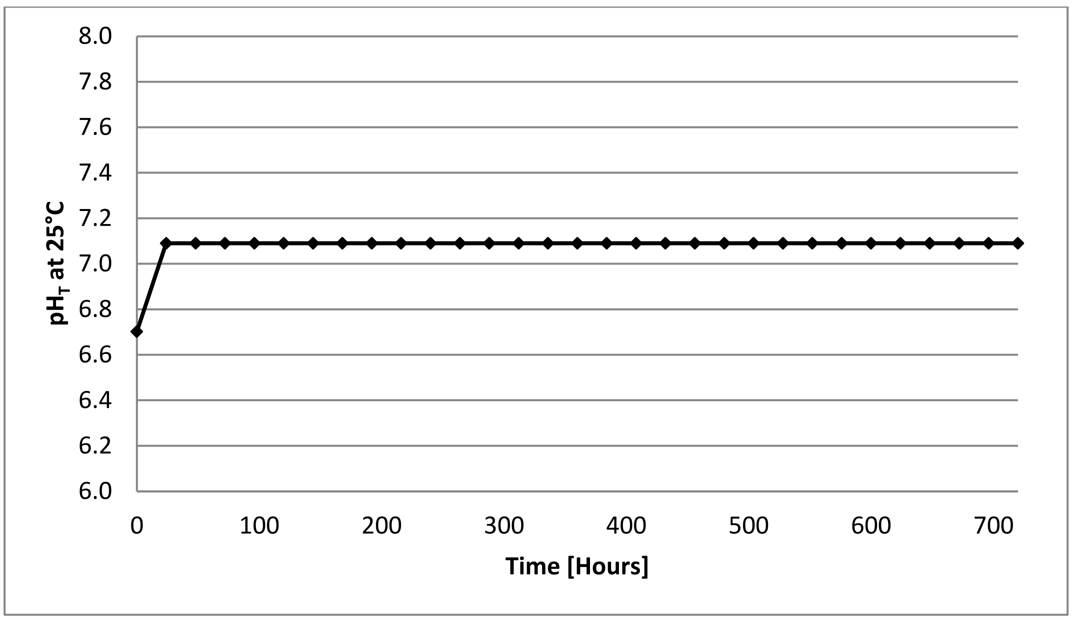

Figure 5-23. The pH_T of the coolant over a 30 day time period

Example temperature sensitivity calculations are performed at 25 degrees C, 100 degrees C, and 200 degrees C across the 30 day time period (see Figure 5-24). At each temperature, the value of neutral pH_T is plotted for comparison.

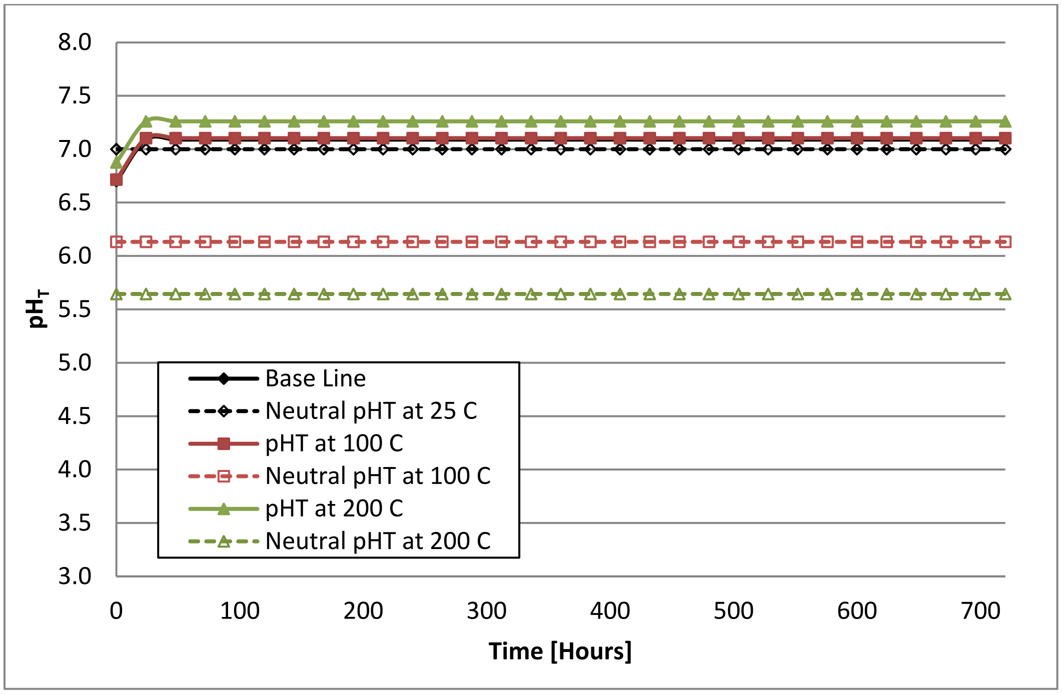

Figure 5-24. Effect of elevated temperature on pH_T

Example boron sensitivity calculations were performed at 1000 ppm, 500 ppm, and 200 ppm (Figure 5-25).

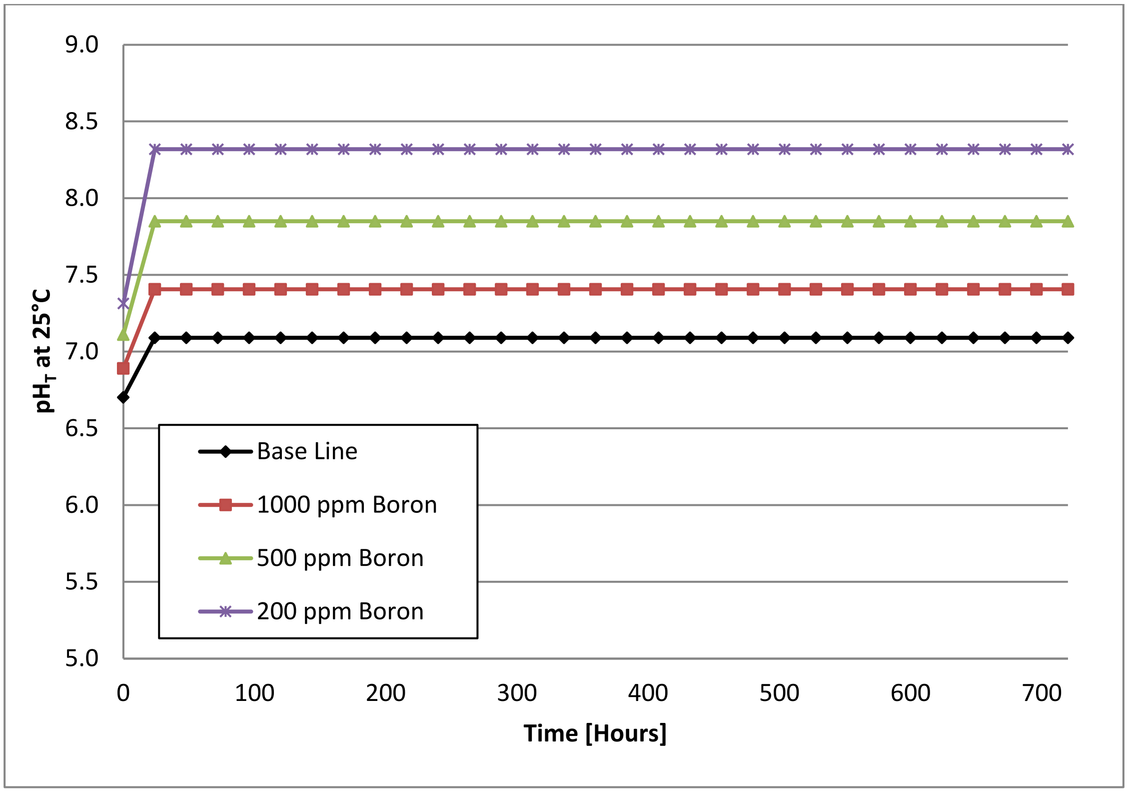

Figure 5-25. Sensitivity of pH_T to boron concentration

Example cesium hydroxide sensitivity analysis was performed at 150 percent and 50 percent of the base line cesium hydroxide (Figure 5-26).

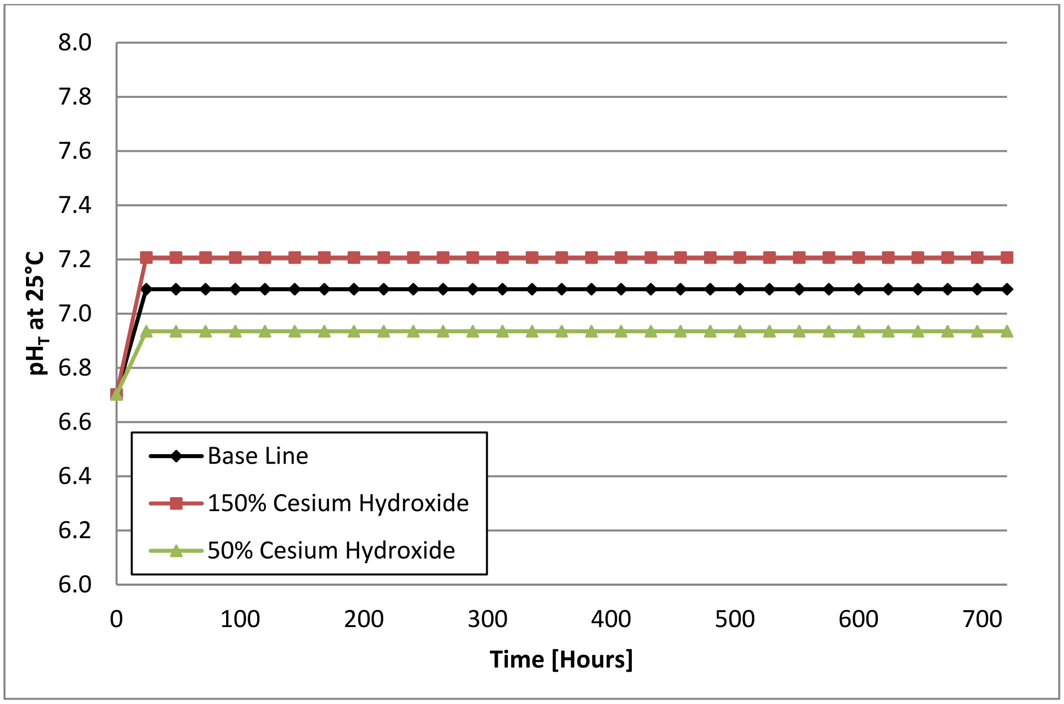

Figure 5-26. Sensitivity of pH_T to cesium hydroxide

The example baseline calculation assumes only nitric acid is generated inside containment. Figure 5-27 evaluates how hydrochloric acid generation would affect pH_T if halogen containing cable insulation was used inside containment.

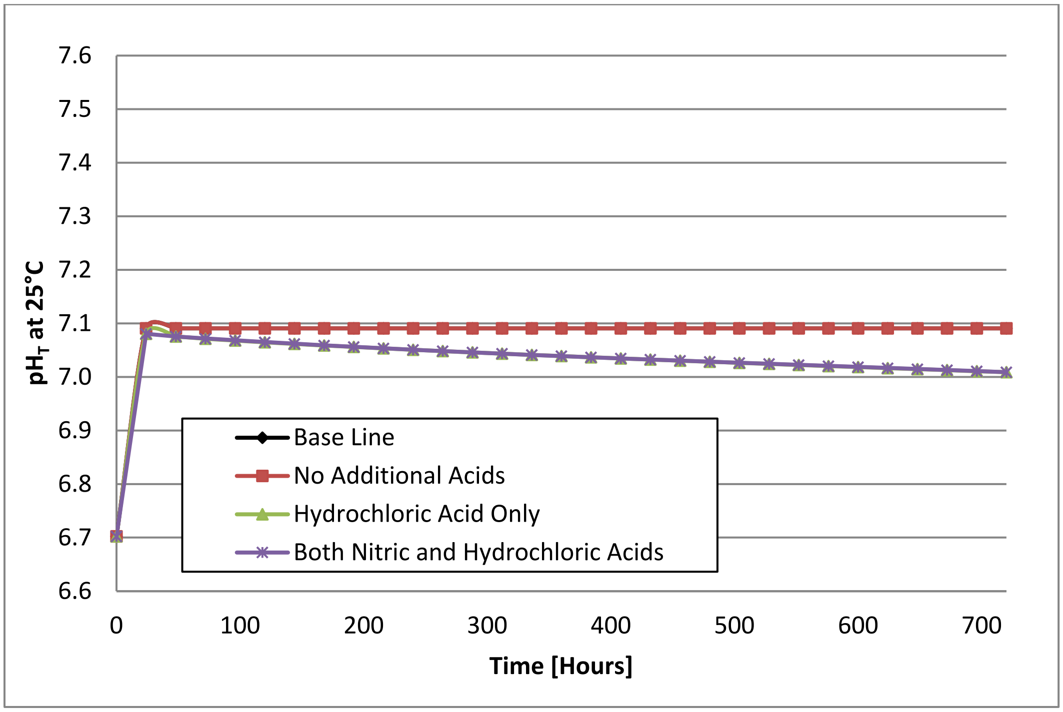

Figure 5-27. Sensitivity of pH$_T$ to nitric acid and hydrochloric acid

Example coolant mass sensitivity was evaluated at a liquid coolant mass of 45,000 kg and 30,000 kg (Figure 5-28).

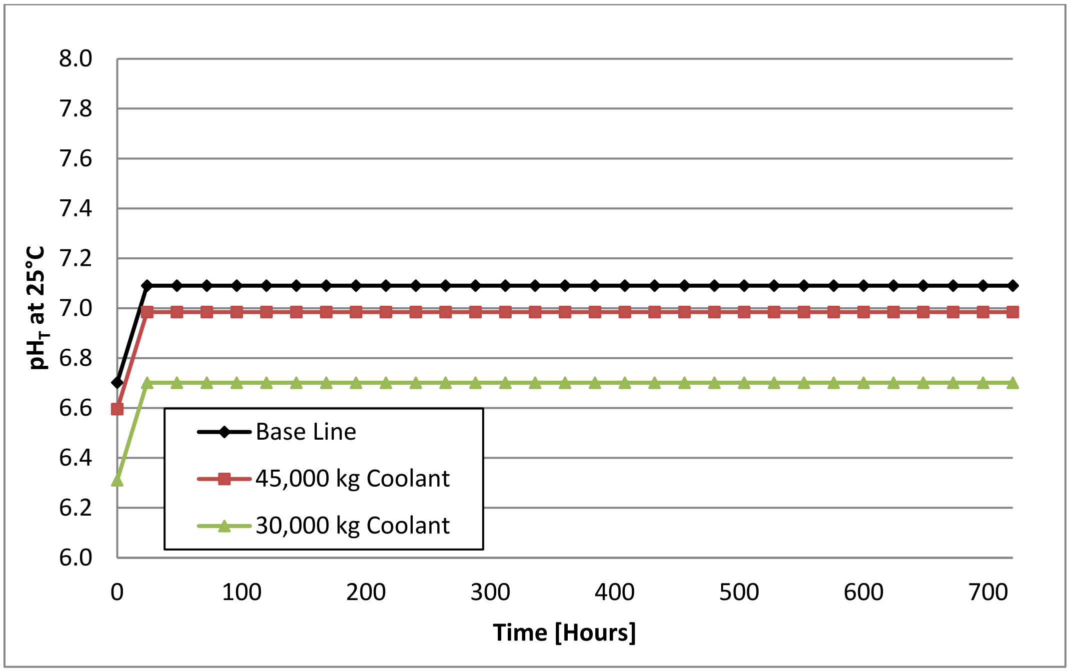

Figure 5-28. Sensitivity of pH_T to the mass of liquid coolant in containment

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 an iodine spike DBST and CDST which together meet 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 CDE radiological consequence analysis associated with the post-accident containment conditions. Consistent with RG 1.183, the assumption that no 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, {{

}}2(a),(c) This insight provides confidence that the aerosol modeling methodology is robust and the inputs utilized for the CDE radiological consequence analysis are conservative.

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 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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3. Ground level release.
4. {{ }}2(a),(c)

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6.1.2 Criteria for Core Radionuclide Inventory

1. {{

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6.1.3 Criteria for Control Room Modeling

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7.0 References

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Appendix A. Regulatory Assessment of Design-Basis and Beyond-Design-Basis Source Terms

The NuScale accident source term methodology distinguishes between design-basis source terms (DBSTs) and the beyond-design-basis core damage source term (CDST). This distinction is made in order to facilitate decoupling the CDST from SSC design and performance requirements for which consideration of a less severe source term is reasonable and appropriate. 10 CFR 52.47(a)(2)(iv) originated as a siting requirement in 10 CFR 100.11, which defined the necessary exclusion area and low population zone for a facility based primarily on the core size and site conditions. TID-14844 (Reference 7.2.33) was developed to loosely encompass the postulated releases evaluated in early license applications. This reflected a realistic appraisal of the consequences of all “significant and credible fission release possibilities,” and yielded a “pipe rupture-meltdown sequence … not likely to be exceeded by any other ‘credible’ accident.”

As reactors became larger, fission product mitigation systems eventually became the dominant factor in meeting the dose limits. Therefore, the core melt source term became the basis for design, performance, and qualification for SSCs relied upon in mitigating fission product releases in order to meet the offsite dose criteria. This requirement is reflected in the definition of “safety related SSC” under 10 CFR 50.2, and related requirements (e.g. 10 CFR 50 Appendix B). In 1997, the requirement was moved to safety analysis report requirements (10 CFR 50.34 et al) to reflect its primary role as the design basis for ESFs.

Following the Three Mile Island accident, the core melt source terms of RGs 1.3 and 1.4 (based on TID-14844) that were developed for siting purposes served as a well-established, surrogate, degraded core event for designing and demonstrating enhanced severe accident mitigation capabilities for operating facilities. This same source term was also utilized for environmental qualification under 10 CFR 50.49, which addresses both qualification of SSCs that are relied on to mitigate offsite radiological consequences from the design-basis core melt MHA (i.e., safety-related functions), as well as enhanced instrumentation requirements implemented to enhance severe accident capabilities following TMI (i.e., nonsafety-related post-accident monitoring instrumentation). Thus, the TID-14844 core melt accident, originally postulated for purposes of a siting evaluation, became a design-basis requirement for numerous SSC design and performance requirements associated with both safety-related and nonsafety-related functions.

Under NuScale’s methodology, the iodine spike DBST is analyzed as a design-basis event and the CDST as a beyond-design-basis event. The design-basis iodine spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv). If both the design-basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met. The analysis of the beyond-design-basis CDST against the acceptance criteria of 10 CFR 52.47(a)(2)(iv) provides reasonable assurance that, even in the extremely unlikely event of a severe accident, the facility’s design features and site characteristics provide adequate protection of the public. The dose analysis for the beyond-design-basis CDST can use assumptions and methods different than those used for design-basis dose analyses if sufficient justification is provided by the application referencing this report.

The classification of CDE as beyond-design-basis is appropriate in order to consistently define the design basis of the plant. Treatment of the CDE as a design-basis accident for some purposes but not others would create an inconsistency in the implementation of regulations. For example, classification of the CDE as a design-basis event for the purposes of the offsite dose evaluation would indicate that SSCs relied upon in meeting the offsite dose limits are performing safety-related functions and subject to environmental qualification for CDE conditions under 10 CFR 50.49. By evaluating both a design-basis and beyond-design basis source term instead of a single design-basis core melt source term, requirements that prescribe the design, quality, and performance of plant features that mitigate the radiological consequences of the respective events are implemented in a manner “commensurate with the importance of the safety functions to be performed,” consistent with GDC 1. Given the use of two source terms to meet 10 CFR 52.47(a)(2)(iv), Table A-1 describes how pertinent requirements are met without a design-basis core damage event.

Table A-1 Summary of Pertinent Requirements

Requirement	Summary	History/Intent	Methodology Compliance
10 CFR 52.47(a)(2)(iv), offsite dose evaluation	Analyze an assumed fission product release into containment to determine offsite radiological consequences are within acceptable limits. Footnote: "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."	Originated as a siting requirement in 10 CFR 100.11, which defined the necessary exclusion area and low population zone for a facility based primarily on the core size and site conditions. As reactors became larger, fission product mitigation systems eventually became the dominant factor in meeting the dose limits. In 1997 the requirement was moved to safety analysis report requirements (10 CFR 50.34 *et al*) to reflect its primary role as ESF design-basis.	The design-basis iodine spike DBST and the beyond-design-basis CDST are each assessed against the radiological criteria of 10 CFR 52.47(a)(2)(iv) to demonstrate that plant features and postulated site parameters limit the offsite radiological consequences of accidents. If both the design-basis iodine spike DBST and the beyond-design-basis CDST analyses show acceptable dose results, 10 CFR 52.47(a)(2)(iv) is met.
General Design Criterion (GDC) 19, control room habitability	"A control room shall be provided from which actions can be taken" under accident conditions. "Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."	Provide reasonable assurance of operator safety while performing actions to mitigate accidents, such that the operator actions are not inhibited. Because the core damage event is normally treated as a DBE with respect to fission product mitigation systems, the core damage event is also normally considered for control room dose to assure operators can perform necessary functions in such an event.	Radiological consequences of design-basis events, including the iodine spike DBST, are assessed against GDC 19. As a beyond-design-basis event, GDC 19 is not directly applicable to the CDST. However, pursuant to 10 CFR 50.34(f)(2)(vii) the GDC 19 design dose criterion is applicable to the CDST, as discussed in the Table A-1 entry for 10 CFR 50.34(f)(2)(vii) below.

Requirement	Summary	History/Intent	Methodology Compliance
10 CFR 50.49, environmental qualification	Design certification applicants must identify, and license applicants must establish a program for qualifying, “electric equipment important to safety,” including safety-related and certain nonsafety-related electric equipment, and “certain post-accident monitoring equipment.” The qualification program must include “the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional.”	Rule development began before, but concluded after, the Three Mile Island (TMI) accident. The “safety-related electric equipment” relied upon for design-basis events includes SSCs (e.g. fission product mitigation ESFs and supporting features) for which the design-basis was the MHA, so they were required to be qualified for the MHA. The rule also includes qualification of “certain post-accident monitoring” (PAM), as addressed by Regulatory Guide (RG) 1.97.	Design-basis events are used to meet 10 CFR 50.49. The beyond-design-basis core damage event is not considered for 10 CFR 50.49. In SECY-90-016 (Reference 7.2.28) NRC Staff acknowledged that the stringent safety-related requirements, including 10 CFR 50.49, were not “commensurate with the importance of the safety functions to be performed” during severe accident mitigation, and recommended relaxed requirements for assuring SSC functionality during a severe accident.

Requirement	Summary	History/Intent	Methodology Compliance
10 CFR 50.34(f)(2)(vii), shielding for vital access and safety equipment 10 CFR 50.34(f)(2)(viii), post-accident sampling	“Perform radiation and shielding design reviews of spaces around systems that may, as a result of an accident, contain accident source term[11] radioactive materials…” “Provide a capability to promptly obtain and analyze samples from the reactor coolant system and containment that may contain accident source term[11] radioactive materials” without exceeding specified worker dose limits.	TMI Items II.B.2 and II.B.3. Per NUREG-0660 (Reference 7.2.35), these Items were short term actions intended to “enhance public safety” by reducing risk from core degradation accidents, which can lead to containment failure and large releases. NUREG-0737 (Reference 7.2.37) specified an assumption of the RG 1.3 (Reference 7.2.46) or RG 1.4 (Reference 7.2.48) source term (based on TID-14844) as the accident conditions for demonstrating these severe accident capabilities.	The CDST is considered as an "accident source term" in addressing these requirements, consistent with the intent to address core degradation accidents. As discussed in NUREG-0737, Item II.B.2, the "adequate access to important areas" provision of 10 CFR 50.34(f)(2)(vii) is satisfied by demonstrating that personnel can perform necessary severe accident mitigation functions--including occupancy of the control room and technical support center--without exceeding the GDC 19 dose design criterion. As a beyond-design-basis event, the control room and technical support center doses may be analyzed with less conservatism than a design-basis analysis, where adequately justified. Protection of necessary equipment from radiation is reasonably assured through demonstrating equipment survivability. The required capability for post-accident sampling or the design dose criterion for obtaining samples may be affected by design-specific features that support an exemption from 10 CFR 50.34(f)(2)(viii).

Requirement	Summary	History/Intent	Methodology Compliance
10 CFR 50.34(f)(2)(xix), core damage monitoring	“Provide instrumentation adequate for monitoring plant conditions following an accident that includes core damage.”	TMI Item II.F.3. The II.F Items aimed to “provide instrumentation to monitor plant variables and systems during and following an accident.” Item II.F.3 concerned instrumentation to support “unplanned action if…a safety system is not functioning” and “action necessary to protect the public and for an estimate of the magnitude of the impending threat.” The requirement is also implemented via RG 1.97 (in addition to 10 CFR 50.49 PAM requirements), which addresses the “expanded ranges” and “damaged core” source term to provide reasonable assurance of the required severe accident capabilities.	In SECY-90-016, staff addressed the issue of assuring that severe accident mitigation features are demonstrated to be available to perform their functions. Staff concluded that severe accident “mitigation features must be designed so there is reasonable assurance that they will operate in the severe-accident environment for which they are intended and over the time span for which they are needed.” Staff concluded that features provided for only severe accident protection were not subject to safety-related requirements such as 10 CFR 50.49 environmental qualification and Appendix B quality assurance. Applications referencing this topical report will provide an equipment survivability evaluation that provides reasonable assurance necessary instrumentation will operate over the time span for which they are needed.

Requirement	Summary	History/Intent	Methodology Compliance
10 CFR 50.34(f)(2)(xxvi), leakage control outside containment 10 CFR 50.34(f)(2)(xxviii), control room leakage	"Provide for leakage control and detection in the design of systems outside containment that contain (or might contain) accident source term[11] radioactive materials following an accident." "Evaluate potential pathways for radioactivity and radiation that may lead to control room habitability problems under accident conditions resulting in an accident source term[11] release, and make necessary design provisions to preclude such problems."	TMI Item III.D.1.1 and III.D.3.4. The III.D Items addressed "design features that will reduce the potential for exposure to workers at nuclear power plants and to offsite populations following an accident." Item III.D.1.1 was a radiological release "source control" measure, that required licensees to reduce leakage to the extent practical "for all systems that could carry radioactive fluid outside of containment," without regard to a particular source term. Item III.D.3.4 was part of worker radiation protection improvements "to allow workers to take effective action to control the course and consequences of an accident." The subsequent rulemaking included a core damage source term in both requirements to address potential new leakage paths from the addition of severe accident mitigation features, and provide reasonable assurance of control room habitability that would be needed to operate those features.	The CDST is considered as an "accident source term" in addressing 10 CFR 50.34(f)(2)(xxvi), consistent with the intent to address core degradation accidents. The assessment of control room radiological consequences for the beyond-design-basis CDE pursuant to 10 CFR 50.34(f)(2)(vii) also satisfies 10 CFR 50.34(f)(2)(xxviii) by considering potential control room leakage pathways.

Requirement	**Summary**	**History/Intent**	**Methodology Compliance**
10 CFR Part 50 Appendix E, paragraph IV.E.8, Technical Support Center (TSC)	A licensee must provide and describe in their emergency plan an “onsite technical support center and an emergency operations facility from which effective direction can be given and effective control can be exercised during an emergency.”	TMI Item III.A.1.2 identified the need for upgraded emergency response facilities (ERFs) to improve the “inadequate” state of emergency planning and preparedness. Specified a TSC as “a place for management and technical personnel to support reactor control functions, to evaluate and diagnose plant conditions, and for a more orderly conduct of emergency operations.” NUREG-0696 (Reference 7.2.62) describes the functional criteria for ERFs, including the TSC, including dose criteria equivalent to the control room.	The design referencing this report will provide a TSC and an evaluation of the radiological consequences of the beyond-design-basis core damage event against 10 CFR 50.34(f)(2)(vii) that provides reasonable assurance that the TSC will be habitable.
Note: The footnote in Table A-1 refers to footnote 11 in 10 CFR 50.34.			

Appendix B. Environmental Qualification Dose Analysis Methodology

This appendix describes the methodology for calculating environmental qualification (EQ) doses in the CNV and bioshield envelope regions. The methodology is for immersion dose rates, photon shine, total integrated radiation doses, and energy deposited for the specified CNV and bioshield envelope regions. {{

}}2(a),(c)

B.1 EQ Dose Methodology Evaluation Scenarios

The goal of this EQ dose methodology is to identify and evaluate a conservative surrogate for the worst-case design basis accident (DBA) for radiation exposures to equipment in the CNV and in the bioshield envelope. The conservative surrogate for the worst-case DBA is identified for each region in the following fashion:

- For equipment in the lower CNV (sump) liquid region – {{

  }}2(a),(c) (Section B.1.1.1)
- For equipment in the upper CNV vapor region – {{

  }}2(a),(c)

  (Section B.1.1.2)
- For equipment in the bioshield envelope – {{

  }}2(a),(c) (Section B.1.2)

Further details of the conservative nature of this EQ dose methodology is provided in the following sections.

B.1.1 Containment Release General Scenario

The nature of a direct primary coolant (plus iodine spike) release to the CNV, as applied in CNV EQ dose evaluations, is conservative. {{

}}2(a),(c)

{{

}}2(a),(c) This treatment of the iodine spike release timing is conservative.

The containment analysis is performed for two separate regions (the upper CNV vapor region and the lower CNV liquid region). {{

}}2(a),(c) This method of defining the CNV regions for either containment

analysis scenario conservatively confines total source inventory to a smaller volume than that of the total CNV free volume.

B.1.1.1 Lower Containment Liquid Region Evaluation Scenario

For the purposes of evaluating the dose to equipment in the lower CNV liquid region, {{

}}2(a),(c)

B.1.1.2 Upper Containment Vapor Region Evaluation Scenario

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}}2(a),(c)

B.1.2 Bioshield Envelope Evaluation Scenario

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}}2(a),(c)

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}}2(a),(c)

The bioshield envelope evaluation scenario described above is conservative {{

}}2(a),(c)

B.2 Assumptions

B.2.1 Activity Plated Out on Containment Surfaces

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}}2(a),(c)

B.2.2 Activity Release Timing

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}}2(a),(c)

B.2.3 Liquid and Vapor RCS Densities

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}}2(a),(c)

B.2.4 Credit for Natural Mechanisms

As stated in Regulatory Guide 1.183, Appendix A, credit may be taken for reduction in the available amount of radiation due to natural deposition mechanisms. {{

}}2(a),(c)

B.2.5 CVCS Purification for Coincident Iodine Spike Calculation

The primary coolant iodine concentration is estimated using a “spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value” (Regulatory Guide 1.183, Appendix E, Item 2.2). {{

}}2(a),(c)

B.2.6 Decay Chain

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}}2(a),(c)

B.2.7 Medium Model

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}}2(a),(c)

B.2.8 Time-Dependent Containment Leak Rate

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}}2(a),(c)

B.3 Methodology

B.3.1 Primary Coolant Source Term

For the EQ dose evaluation, the primary coolant radionuclide inventory described in Section 3.3.2 of this report, including isotopic concentrations equivalent to the design basis DE I-131 and DE Xe-133 limits, is applied.

B.3.1.1 Non-Iodine Spiking

Spiking of radionuclides besides iodine is not explicitly considered in this methodology. This approach is consistent with the available regulatory guidance, which does not prescribe the spiking of radionuclides besides iodines. Regulatory Guide 1.183, Appendix I, addresses assumptions for evaluating radiation doses for equipment qualification purposes. Regulatory Position 4 therein notes the possibility that “another design basis accident” (i.e., non-core melt events) may be more limiting than the "design basis LOCA" (i.e., the core melt source term event) for the purposes of equipment qualification for some components. In these cases, RG 1.183 recommends the use of the applicable assumptions of Appendices B through H otherwise applicable to the dose consequence evaluations for the event in question, which do not include spiking of any radionuclides other than iodines. A key use of the iodine spike DBST is to establish the radiation environment for a design basis accident inside containment, as other events are expected to be more limiting with respect to dose consequences outside containment. Therefore, NuScale concludes the existing guidance of RG 1.183, which establishes the assumptions of Appendices B through H are adequate for a similar use (developing design-basis EQ doses), is appropriate to follow with respect to the iodine spike DBST. {{

}}2(a),(c)

B.3.1.2 Coincident Iodine Spiking

The coincident iodine spike modeling approach used in this methodology {{
}}2(a),(c)

B.3.2 Energy, Dose Rates, and Integrated Dose

The total energy rate for a given isotope is based upon its initial activity and average energy per decay. To calculate the activity of an isotope sometime after shutdown, a standard exponential decay model is used to extrapolate the values based on isotopic half-lives, as described by Eq. B-1.

$$A_i(t) = A_{o,i} e^{\frac{-ln|2|t}{T_{1/2}}} \qquad \text{Eq. B-1}$$

where,

$A_i\,(t)$ = Activity of isotope i at time t, Ci

$A_{O,i}$ = Initial activity of isotope i, Ci

$T_{1/2}$ = Half-life for isotope =i, s

t = time at which to calculate the activities, s

{{

}}2(a),(c) With activities determined for a given hourly interval, photon or electron energy emission rate in units of MeV/s are calculated based on the average photon or electron emission rate for a single disintegration, or nuclear transformation. The unit “nt”, an abbreviation for nuclear transformation, is used. This “nt” unit is equal to one becquerel (Bq). Multiplying an activity, “A”, by the average energy, “E”, results in the energy emission rate, “R”, given by

$$[A]Ci \cdot \frac{3.7E10\frac{nt}{s}}{Ci} \cdot [E]\frac{MeV}{nt} \rightarrow [R]\frac{MeV}{s} \qquad \text{Eq. B-3}$$

Finally, the dose rate in units of rad/hr may be calculated based on the energy emission rate, volume and density of interest, and several unit conversions as expressed by

$$[R]\frac{MeV}{s}\cdot\frac{3600\,s}{hr}\cdot\frac{1.602\,E-19\,J}{eV}\cdot\frac{10^6}{M}\cdot\frac{Rad}{0.01\,\frac{J}{kg}}\cdot\frac{1}{[V]\,ft^3}\cdot\frac{ft^3}{[\rho]\,\mathrm{lbm}}\cdot\frac{2.2046\,lbm}{kg}\rightarrow\frac{Rad}{hr} \qquad \text{Eq. B-4}$$

The total photon emission energy rate or dose rate is then the sum of all the emission or dose rates for all the isotopes considered, as is the case for the total electron rates. At each time step, the rates and integrated emitted energy or dose may be calculated. The integrated energy emitted in MeV is calculated for the example 2400 hour duration as follows

$$Energy\ Emitted\ (MeV) = \sum_{i=0}^{2400}(Energy\ Rate)_i\left(\frac{MeV}{s}\right)\cdot 3600\left(\frac{s}{hr}\right)\cdot 1\ hr \qquad \text{Eq. B-5}$$

Similarly, the integrated dose in units of rad is given by

$$Dose\ (Rad) = \sum_{i=0}^{2400}(Dose\ Rate)_i\left(\frac{Rad}{hr}\right)\cdot 1\ hr \qquad \text{Eq. B-6}$$

{{

}}2(a),(c)

B.3.3 Containment Leakage

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}}2(a),(c)

{{

}}2(a),(c)

B.4 Summary and Conclusions

In summary, a methodology for calculating EQ doses is described. Notable conservatisms of this methodology include:

- {{

}}2(a),(c)

Section C

RAI Number	eRAI Number	NuScale Letter Number
8800	8800	RAIO-0717-55010
8881	8881	RAIO-0817-55640
9224	9224	RAIO-0718-61146
9646	9646	RAIO-0219-64724
9224	9224	RAIO-0419-65259
9690	9690	RAIO-0719-66517
9690	9690	RAIO-0919-66890

July 24, 2017

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 8800 (eRAI No. 8800) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 1

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 8800 (eRAI No. 8800)," dated June 23, 2017
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 1, dated April 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 8800:

- 15.00.03-2
- 15.00.03-3
- 15.00.03-4

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 8800 (eRAI No. 8800). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, TWFN-6E55
Samuel Lee, NRC, TWFN-6C20
Anthony Markley, NRC, TWFN-6E55

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8800, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8800, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0717-55012

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8800, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8800
Date of RAI Issue: 06/23/2017

NRC Question No.: 15.00.03-2

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.1.3 Regulatory Guide 1.183 provides dose assessment guidance.

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

The proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is an extrapolation of an equation from the Burley paper ("Evaluation of Fission Product Release and Transport for a Fuel Handling Accident," G. Burley, NRC, Oct. 5, 1997) that forms the underlying basis for the pool iodine decontamination factor given in RG 1.183. As stated on page 16 of the Burley paper, the most important parameters related to the iodine decontamination factor within the pool include the gas bubble dimensions, contact time and partition factor.

a. The methodology proposed assumes that the range of gas bubble characteristics (e.g., bubble diameter, bubble effective diameter, bubble velocity) is not different at rise heights over 23 feet. Please provide justification for this assumption.
b. The Burley paper assumed that the time for contact between the pool water and the gas bubbles as they rise to the surface of the pool was short

compared to the time it takes to get to an equilibrium iodine concentration. How did you determine if this assumption is applicable to the deeper pool depth for the NuScale design?

c. What is the basis for applying the partition factor ranges used in the Burley paper to bubble rise heights greater than 23 feet?

NuScale Response:

Question a.

The proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is withdrawn and is replaced with the standard Regulatory Guide 1.183 Appendix B guidance of using an overall effective decontamination factor of 200 when the depth of water above the damaged fuel is 23 feet or greater. Therefore, no justification for the assumption of gas bubble characteristics associated with the previously proposed methodology is provided.

A markup of TR-0915-17565 is provided to show this methodology change.

Question b.

As discussed in the response to Question a., the proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is withdrawn. Therefore, no justification for the assumption of gas bubble contact time associated with the previously proposed methodology is provided.

Question c.

As discussed in the response to Question a., the proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is withdrawn. Therefore, the basis for applying the partition factor ranges used in the Burley paper to bubble rise heights greater than 23 feet is not provided.

Impact on Topical Report:
Topical Report TR-0915-17565, Accident Source Term Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

TABLES

7. STARNAUA is appropriate for modeling natural removal of containment aerosols for the NuScale design.
8. No maximum limit on ~~elemental~~ iodine decontamination factor for natural removal of containment aerosols.
9. {{ }}2(a),(c)
10. Utilizing the iodine spiking assumptions of RG 1.183 is appropriate.
11. ~~Generalized process for determining the analytical effective decontamination factor based on a minimum depth of water above the damaged fuel in a fuel handling accident.~~Utilizing the iodine decontamination factor assumptions of RG 1.183 for the fuel handling accident is appropriate.
12. With respect to accident analysis, it is appropriate to neglect the small secondary side volume that could contain activity from primary to secondary leakage for the NuScale design.
13. For pH_T values of 6.0 or greater, the amount of iodine re-evolution that could occur between pH_T values of 6.0 and 7.0 is negligible and not included in the dose calculation.
14. Containment shine of the radiation in the containment airspace through the containment vessel, reactor pool water, and then through the reactor building walls or ceiling to the environment is negligible for the NuScale design.

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

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, as updated by Regulatory Issue Summary (RIS) 2006-04 (Reference 7.2.11) item 8, 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 all the fuel rods in one irradiated fuel assembly 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
- overall effective iodine decontamination factor of 200 for the pool
- ~~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

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. An elemental decontamination factor of 285, an organic decontamination factor of 1, and an overall effective decontamination factor of 200 are assumed per RG 1.183 as updated by RIS 2006-04 (Reference 7.2.11) item 8.~~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~~

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

~~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~~

~~{{~~

~~}}~~

{{

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

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

{{

}}2(a),(c)

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.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8800
Date of RAI Issue: 06/23/2017

NRC Question No.: 15.00.03-3

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

On page 33 of the topical report, it states that based on holding all parameters other than depth of water above the fuel fixed, the inorganic iodine decontamination factor is scaled (from 285, as given in RG 1.183 for 23 ft) by a proprietary factor that includes consideration of the water depth. Please provide the derivation of this scaling factor.

NuScale Response:

As discussed in the response to RAI Question 15.00.03-2 a., the proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is withdrawn. Therefore, the derivation of the scaling factor associated with the previously proposed methodology is not provided.

Impact on Topical Report:
There are no impacts to the Topical Report TR-0915-17565, Accident Source Term Methodology, as a result of this response.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8800
Date of RAI Issue: 06/23/2017

NRC Question No.: 15.00.03-4

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses, 10 CFR Part 50, Appendix A, GDC 19 for control room radiological habitability, and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

NuScale licensing topical report TR-0915-17565-P, Rev.1, "Accident Source Term Methodology," provides a proposed methodology for the performance of design basis accident radiological consequence analyses for the NuScale design. The staff requires the following information to complete its review of the subject topical report to evaluate compliance with the applicable NRC requirements:

The proposed method to determine the iodine effective DF also provides for a sensitivity study based on the inorganic fraction of fuel rod gap iodine assumed to be released to the pool. What is the basis for the implied assumption that the inorganic iodine fraction released from the fuel rod gap in the fuel handling accident would be different for the NuScale fuel than the value given in RG 1.183?

NuScale Response:

As discussed in the response to RAI Question 15.00.03-2 a., the proposed methodology for determining the iodine decontamination factor for the pool during a fuel handling accident is withdrawn. Therefore, no discussion of the sensitivity study or implied assumptions associated with the previously proposed methodology is provided.

Impact on Topical Report:
There are no impacts to the Topical Report TR-0915-17565, Accident Source Term Methodology, as a result of this response.

NuScale Power

RAIO-0817-55640

August 24, 2017

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 8881 (eRAI No. 8881) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 1

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 8881 (eRAI No. 8881)," dated July 25, 2017
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 1, dated April 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 8881:

- 02.03.04-2

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 8881 (eRAI No. 8881). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Darrell Gardner at 980-349-4829 or at dgardner@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Anthony Markley, NRC, OWFN-8G9A

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 8881, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 8881, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0817-55644

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 8881, nonproprietary

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Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 8881
Date of RAI Issue: 07/25/2017

NRC Question No.: 02.03.04-2

Regulatory Background

10 CFR 52.47(a)(1) requires a DC applicant to provide site parameters postulated for the design and an analysis and evaluation of the design in terms of those site parameters. 10 CFR 52.47(a)(2)(iv) requires a DC applicant to perform an assessment of the plant design features intended to mitigate the radiological consequences of accidents, which includes consideration of postulated site meteorology, to evaluate the offsite radiological consequences at the exclusion area boundary (EAB) and outer boundary of the low population zone (LPZ). Regulatory Guide (RG) 1.145, "Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants," presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of radiological releases to the EAB and outer boundary of the LPZ.

Information Request Background

One of the positions that NuScale is seeking approval for in TR-0915-17565-P is the use of the ARCON96 methodology for the calculation of offsite atmospheric dispersion factors (or X/Q values). ARCON96 has typically been used to calculate X/Q values for the control room based on guidance provided in RG 1.194, “Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants.” Section 4.1.6 of TR-0915-17565-P describes the methodology to be used in utilizing ARCON96 in performing offsite atmospheric dispersion calculations.

Supplemental to RAI 8691, Question 02.03.04-1

In RAI 8691, Question 02.03.04-1, the NRC staff asked why it is acceptable for the NuScale design-basis accident offsite atmospheric dispersion methodology to (1) use the 95th percentile X/Q value as the maximum sector X/Q value instead of a 99.5 percentile X/Q value as suggested by RG 1.145 and (2) {{

}}[2(a),(c)] as suggested by RG 1.145.

In response to RAI 8691, Question 02.03.04-1, the applicant stated that it is acceptable to utilize the 95th percentile X/Q value as the maximum sector X/Q because RG 1.194 directs that the 95th percentile X/Q value should be determined for control room related atmospheric dispersion analyses. According to the applicant, the use of ARCON96 (RG 1.194) methodology versus the PAVAN (RG 1.145) methodology creates non-analogous situations when trying to combine guidance from both RG 1.194 and RG 1.145 or compare NuScale's methodology to these regulatory guides in calculating offsite X/Q values.

The applicant also stated that {{

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

In response to the applicant's statements, the NRC staff finds {{
}}[2(a),(c)] of RG 1.145 for the EAB and outer boundary of the LPZ and the 95 percentile X/Q approach of RG 1.194 for the control room to be analogous. Both approaches state that the X/Q value that is not exceeded by more than 5 percent of the X/Q values generated with all the meteorological observations should be used to estimate doses to the appropriate receptor.

One of the differences between the RG 1.145 (PAVAN) and RG 1.194 (ARCON96) applications is the nature of the receptors. Each ARCON96 run simulates a release to a single receptor point, such as an air intake or infiltration pathway. In contrast, each PAVAN run simulates a release to a boundary such as the EAB and/or the outer boundary of the LPZ. 10 CFR 52.47(a)(2)(iv) states that the contents of applications for standard design certifications should contain an evaluation and analysis of a postulated fission product release demonstrating that an individual located at any point on the boundary of the EAB for any 2-hour period and an individual located at any point of the outer boundary of the LPZ during the entire period of plume passage would not receive a radiation dose in excess of 25 rem total effective dose equivalent (TEDE). {{
}}[2(a),(c)] of the EAB and LPZ boundaries for the duration of the dose assessment. Therefore, the staff's suggestion {{
}}[2(a),(c)] for the EAB and outer boundary of the LPZ is reasonable and consistent with current regulatory guidance.

RG 1.145 also implements a direction-dependent methodology as described in NUREG/CR-2260. The RG 1.145 direction-dependent methodology considers the directional variability of diffusion conditions and boundary distances by dividing the 360-degree EAB and outer boundary of the LPZ into sixteen 22.5-degree sectors. If atmospheric dispersion conditions and boundary distances are identical in each of the sixteen 22.5-degree wind direction sectors, then the calculated direction-independent 5 percent X/Q value would be exceeded approximately 0.31 percent of the time in each of the 22.5-degree downwind sectors (5% times 22.5°/360°). After a parametric study of dispersion conditions at a number of sites, NUREG/CR-2260 concludes that the 0.5 percent level for the direction-dependent methodology would be reasonably consistent with the direction-independent 5 percent approach.

NuScale suggests that a 95th percentile X/Q value be calculated for each of sixteen 90-degee sectors and the maximum 95th percentile X/Q concentration out of all 16 sectors be selected. If atmospheric dispersion conditions and boundary distances are identical in each of the sixteen 90-degree wind direction sectors, then the calculated direction-independent 5 percent X/Q value would be exceeded approximately 1.25 percent of the time in each of the 90-degree downwind sectors (5% times 90°/360°). Therefore, NuScale's suggested use of the 5 percent X/Q value for the 90-degree sector X/Q is less conservative than the 1.25 percent 90-degree sector X/Q value implied by the NUREG/CR-2260 methodology.

Given the above discussion, the applicant should provide additional justification for deviating from RG 1.145 guidance regarding why it is acceptable to (1) use the 95th percentile 90- degree X/Q value as the maximum sector X/Q value instead of the 99.5 percentile 22.5- degree X/Q value and (2) {{ }}[2(a),(c)] Alternatively, as discussed during the closed meeting on May 23, 2017, {{ }}[2(a),(c)]

NuScale Response:

The offsite atmospheric dispersion methodology is modified to {{ }}[2(a),(c)]

A markup of TR-0915-17565 is provided to show this methodology change.

Impact on Topical Report:
Topical Report TR-0915-17565, Accident Source Term Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

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~~NARCON

The calculation of both onsite and offsite atmospheric dispersion factors for design basis accidents is performed with NARCON~~96 (Reference 7.2.24)~~. NARCON is the NuScale version of ARCON96 (Reference 7.2.24). NARCON is equivalent to ARCON96 with the exceptions of input/output edit differences {{

}}

{{

}}2(a),(c)~~only~~. ~~The program~~ARCON96 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

[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~~overall 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

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

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

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. {{

}}2(a),(c)

- 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~~{{

  }}2(a),(c).

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. {{

}}2(a),(c)

~~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.

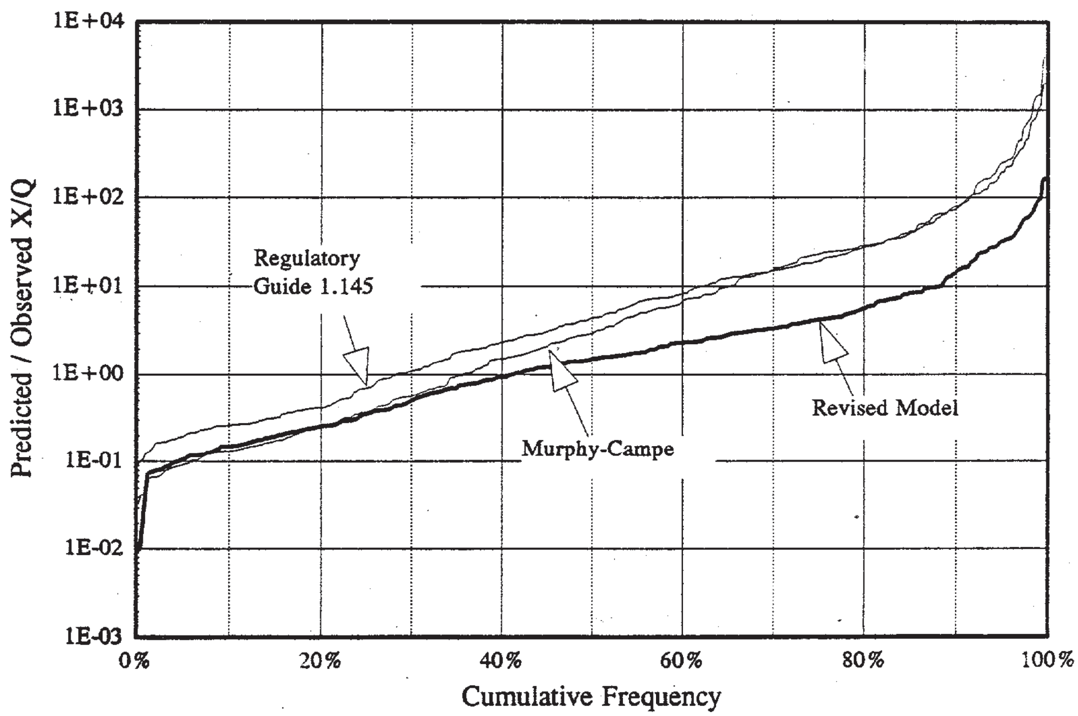


Figure 4-1. Cumulative frequency distributions of predicted to observed concentration ratios for the Murphy-Campe (RG 1.145), and revised models (Reference 7.2.23)

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

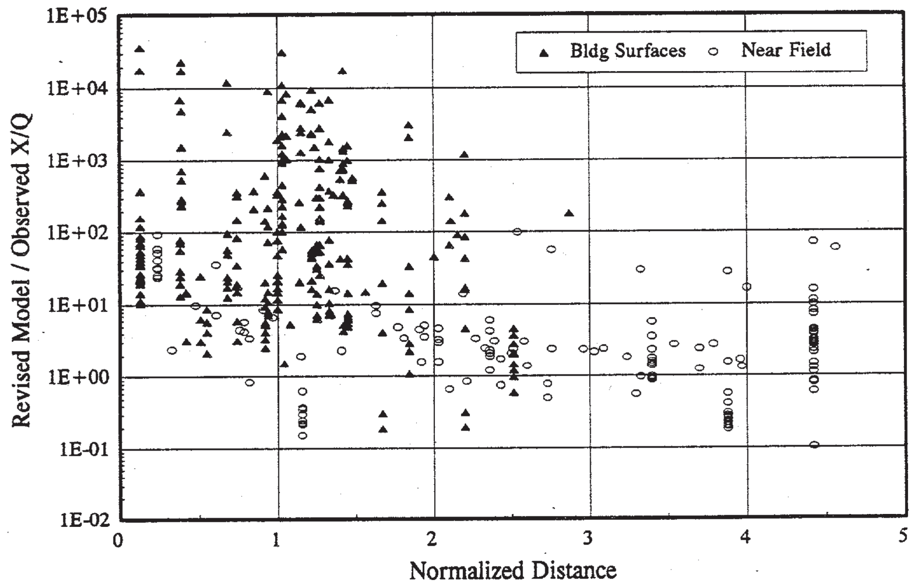


Figure 4-6. Ratios of predicted to observed concentrations for ARCON96 (Reference 7.2.23)

{{

}}2(a),(c)

4.1.5.2 Test Case Two: Distance Comparison

{{

}}2(a),(c)

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

{{

}}2(a),(c)

{{

}}2(a),(c)

Table 4-3. Site meteorological statistics

{{

}}2(a),(c)

4.1.6 Application

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

- For each possible measured wind direction sector available in the input meteorological data (typically 16 sectors), {{

  }}2(a),(c)

- Ground level release (no credit taken for possible elevated release)
- {{

  }}2(a),(c)

- {{

}}2(a),(c)

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. Final~~NuScale plans to provide the final~~ design values are provided as part of~~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. All examples provided in Section 5 are based on a superseded preliminary version of the NuScale design. Since the purpose of these example results is illustrative and the changes in results would not be large enough to provide new insights into the application of the methodologies, the example results are not updated as methodology changes and revisions to this report occur. Differences in the methodologies originally utilized to create these example results and the methodologies stated in the current revision of this report are noted in the associated Section 5 subsections as appropriate.

5.1 Atmospheric Dispersion Factors

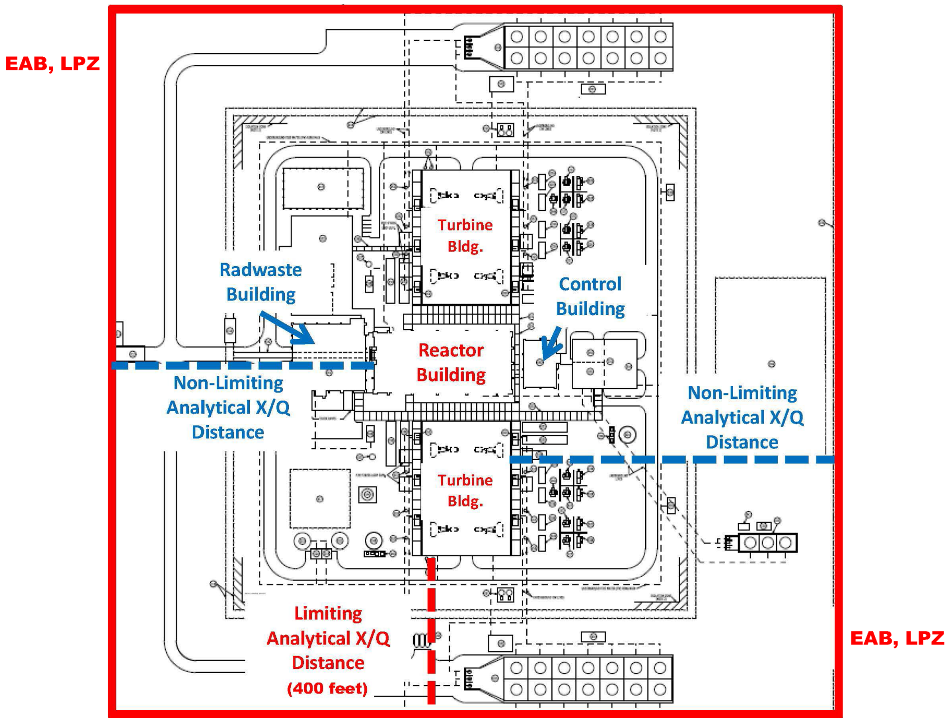
~~{{~~ A COL applicant that implements this methodology is expected to use site-specific atmospheric dispersion factors calculated from qualified site-specific meteorological data obtained from a site specific RG 1.23 compliant meteorological monitoring program. In order to demonstrate the application of this methodology, this report assumed a three year data span for Sacramento, California from 1984 to 1986 in example calculations, described in Section 5.2 and Section 5.3. This representative site is assumed to occur on flat ground with nominal surface features (i.e., default surface roughness). A COL applicant who utilizes this methodology is expected to evaluate the applicability of the atmospheric dispersion modeling methodology for any significant site-specific geographical features. The example site information evaluated in this section is a representative example and is intended to illustrate how dispersion factors are calculated utilizing the methodology from Section 4.1 (with the exceptions of {{

}}$^{2(a),(c)}$) as applied to a set of U.S. meteorological data.

~~In order to utilize appropriate dispersion factors for design certification,~~In order to establish an appropriate site and associated meteorological dataset that could be used to develop atmospheric dispersion factors in a design certification, the methodology from Section 4.1 (with the exceptions of {{

}}$^{2(a),(c)}$) is applied to a set of U.S. meteorological data from 241 sites across the U.S. from which a site representative of an 80-90th percentile U.S. site was selected; as recommended in the advanced light water reactor (ALWR) utility requirements document (URD), Reference 7.2.42 (which specifically recommended 80-90th percentile). The selected site meteorological data is then used in example calculations of offsite and control room atmospheric relative concentration values.

{{



}}2(a),(c)

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

{{ The example analysis assumed a conservative cross sectional building area of 0.01 square meters, since smaller cross sectional building areas have been observed to produce larger relative concentrations. Note that ARCON96 has only one input for cross sectional building area and therefore this input accounts for the effect of all buildings between the source and receptor. All source geometries were assumed to be from a ground-level point source; no elevated, vent, heated, or diffuse sources were considered. The site terrain elevation differences were assumed to be zero.

ARCON96 was executed using data from each of the geographical sites in the selected meteorological database and executed 16 times for each site; once for each direction sector. The sectors are centered at ~22.5 degree intervals~~,~~ {{

}}2(a),(c) and each is 90 degrees in width,

{{

}}2(a),(c). The maximum relative concentrations were selected for each site at each time period and distance. A set of 80th percentile relative concentrations and a set of 90th percentile relative concentrations were established by ordering the data from least to greatest and selecting the 80th and 90th percentile data points for each distance and each time period. Selection of an 80-90th percentile site is based on establishing a site whose relative concentrations typically fall between the 80th and 90th percentile relative concentration data sets. }}2(a),(c)

~~{{~~ Table 5-2 presents the ratio of the selected relative concentration to the true 90th percentile relative concentration in the dataset. Though not all values represent the 80-90th percentile of the dataset, all of them are reasonably close in magnitude. There are five values below the 90th percentile of the dataset, but all of them are close to the 90th percentile in magnitude. Considering this relationship, and the fact that many of the selected concentrations are well above the 90th percentile, the selected site is justified for use as the 80-90th percentile of the dataset. ~~}}2(a),(c)~~

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

{{

Downwind Distance (m)	0-2 hour	2-8 hour	8-24 hour	1-4 day	4-30 day
33	1.00	1.02	0.88	0.98	1.08
66	1.01	0.99	0.88	0.98	1.07
122	1.00	1.01	0.88	0.99	1.08
201	1.00	1.02	0.88	0.99	1.06
402	1.01	1.00	0.92	0.96	1.03
805	0.99	0.93	1.15	1.06	1.06
1609	0.90	0.96	1.25	1.08	1.13

Using the methodology specified in Section 4.1 of this report (with the exceptions of {{

}}2(a),(c), offsite atmospheric relative concentration values for the site located in Sacramento, California for the data span of three years (1984 to 1986) were calculated. These relative concentrations (shown in Table 5-4) are used in example dose calculations in this report. A COL applicant is expected to use site-specific atmospheric dispersion factors calculated from qualified site-specific meteorological data obtained from a site specific RG 1.23 compliant meteorological monitoring program. Use of NWS data is only for the purpose of illustrating the derivation of reasonable atmospheric relative concentration values for the NuScale design certification application and example calculations in this report. ~~}}2(a),(c)~~

Table 5-3. Selected meteorological data

{{

WBAN	Location	Number of Years	Span of Years
23232	Sacramento, California	3	1984-1986

The calculated offsite dispersion factors are presented in Table 3-2. ~~}}2(a),(c)~~

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

{{

Distance (feet)	Distance (meters)	0-2 hour (s/m³)	2-8 hour (s/m³)	8-24 hour (s/m³)	1-4 day (s/m³)	4-30 day (s/m³)
400	121.9	5.72E-04	4.85E-04	2.14E-04	2.15E-04	1.95E-04

}}2(a),(c)

5.1.2 Control Room and Technical Support Center Dispersion Factors

{{ Possible reactor or turbine building source locations, including doors, heating, ventilation, and air conditioning (HVAC) inlets and outlets, and penetrations, were examined for determining the limiting source locations. For the control room envelope and technical support center, personnel access doors and HVAC inlets were examined as possible receptor locations. In these example calculations, the control room ventilation air exhaust was not included as a control room receptor, because it was assumed that the control room emergency air will be continuously discharged through this location.

Utilizing the three dimensional coordinates provided by building drawings, the total and horizontal distances between source and receptor were calculated for each source-receptor combination. The total "taut-string" distance was considered as a vector length, therefore the standard equation for calculating vector lengths was utilized. The resultant control room atmospheric dispersion factors are presented in Table 5-5 for the limiting control room source-receptor distance. }}2(a),(c)

Table 5-5. Example control room atmospheric dispersion factors

{{

Distance (feet)	Distance (meters)	0-2 hour (s/m³)	2-8 hour (s/m³)	8-24 hour (s/m³)	1-4 day (s/m³)	4-30 day (s/m³)
111.78	34.1	6.27E-03	5.37E-03	2.31E-03	2.35E-03	2.13E-03

}}2(a),(c)

5.2 Category 1 Events

{{ Example dose results from the Category 1 events described in Section 3.2 of this report are shown in Table 5-6. The acceptance criteria in Table 5-6 are taken from SRP Section 15.0.3. These example calculations utilized the dispersion factors associated with the limiting 80th-90th percentile site described in Section 5.1, and the general methodologies described in Section 3.3 (with the exception of the method utilized to calculate the iodine decontamination factor for the pool during a fuel handling accident), as applied to the example design assumed in these evaluations. For this example, the smallest margin between calculated and acceptance criteria dose is a factor of 7 smaller between the 5 Roentgen equivalent man (rem) control room dose acceptance criteria and the 0.72 rem calculated value.

A sensitivity study was performed for the SGTF and MSLB events assuming the liquid secondary coolant in the steam generator was at the primary coolant design basis limit concentration. This study resulted in an EAB dose increase of 1.4E-03 rem TEDE, as compared to the acceptance criteria of 2.5 rem or 25 rem, depending on the iodine

6.1.1 Criteria for Atmospheric Dispersion Factors

1. {{
2. }}2(a),(c)
3. Ground level release.
4. {{
5.
6. }}2(a),(c)

6.1.2 Criteria for Core Radionuclide Inventory

1. {{
2.
3.
4.
5. }}2(a),(c)

6.1.3 Criteria for Control Room Modeling

1. {{
2. }}2(a),(c)

July 30, 2018

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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One White Flint North
11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9224 (eRAI No. 9224) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9224 (eRAI No. 9224)," dated December 04, 2017
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2, dated April 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 9224:

- 01.05-32

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9224 (eRAI No. 9224). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Jennie Wike
Manager, Licensing
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8G9A
Samuel Lee, NRC, OWFN-8G9A
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9224, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

Enclosure 3: Affidavit of Thomas A. Bergman, AF-0718-61147

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

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NUSCALE
Power for all humankind

Response to Request for Additional Information

eRAI No.: 9224
Date of RAI Issue: 12/04/2017

NRC Question No.: 01.05-32

Regulatory basis

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability; and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Request for additional information

In the NRC staff's review of the topical report (TR), "Accident Source Term Methodology," TR-0915- 17565-P, Rev. 2, the staff requires the following information to complete its review. Also, the staff requests that the requested information be included in the TR, as appropriate.

1. The proposed methodology for determining the design basis source term in Section 4.2.1 includes using the median release fraction from the MELCOR calculations for the more likely severe accident scenarios. The NuScale Final Safety Analysis Report (FSAR) implements the methodology by taking the median over four MELCOR calculations. What is the basis for using the median release fraction as opposed to another statistical metric such as a mean or a 75th percentile? Given the limited number of scenarios (four) as implemented in the FSAR, what is the basis for using a median release fraction to evaluate the offsite radiological consequences as opposed to evaluating the offsite radiological consequences for each of the four scenarios?

2. The second paragraph of Section 4.2.4 describes the methodology {{

}}2(a),(c)

{{ }}$^{2(a),(c)}$ However, the methodology is not clear to the staff. The applicant is requested to clarify {{

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

3. Enclosure 1 to NuScale letter to NRC dated February 6, 2017, provided clarification of the basis of its use of an inert species ratio (i.e., ratio of non-radioactive aerosol to radioactive aerosol) of 2 to 4. As part of the clarification, NuScale stated that past studies showed an inert species ratio of 2 to 4 and that the NuScale fuel design is based on a conventional PWR fuel design with similar geometries, materials, and fuel composition with a similar ratio of control material to fission products. The staff's subsequent review of additional NuScale documents as part of the design certification review suggests that the inert species ratio could be different from that seen in the past studies because NuScale's core is different from previous PWR designs in two ways – it does not use 1% tin in its cladding and it mainly uses B4C control rods. The applicant is requested to provide additional information supporting the use of a ratio of 2 to 4 given these two differences.

4. With respect to the example implementation of the methodology on aerosol removal, the two curves (aerosol concentration and aerosol removal rate constant) in Figure 5-13 of the TR appear to be inconsistent with each other. For the period beginning at the end of the release, the decrease in the aerosol concentration has a lower removal rate than shown in the removal rate constant curve. Please clarify this apparent inconsistency and how the curves were calculated.

NuScale Response:

1. NuScale now utilizes the term "core damage maximum hypothetical accident (MHA)" instead of the term "design basis source term".

It should be noted that the number of severe accident scenarios implemented in determining the core damage MHA had been updated from four to five, as detailed in the previously submitted May 24, 2017 response to RAI 8774, Question 15.00.03-1 (ML17144A451). The example core damage MHA analysis shown in TR-0915-17565-P, Rev. 3 will be updated to use a set of five severe accident scenarios.

As noted in Section 4.2.3 of TR-0915-17565-P, Rev. 2, the use of the median is similar to the approach used in Sandia National Laboratory report SAND2011-0128 (Reference 7.2.10 of TR-0915-17565-P, Rev. 2). SAND2011-0128 gives the following justification for use of the median release fraction:

“The median is taken to be the representative value of the source term distribution… A percentile other than the median as the representative magnitude would require justification from some other source. By adopting the median, half the accidents have larger release fractions and longer release times than the representative accident and half have smaller release fractions and shorter release times.”

While SAND2011-0128 notes as a limitation that "results are not applicable to small modular reactors that could have accident processes that differ substantially from those of the large power plants considered here", this limitation applies to results rather than evaluation methods. It is NuScale’s view that the SAND2011-0128 release fraction selection method represents an applicable precedent of a solution approach to the problem of selecting representative release fractions from multiple postulated severe accident scenarios.

It is further noted that the implemented median release fractions from the MELCOR severe accident scenarios in the example core damage MHA analysis are conservatively higher than the mean release fractions from the MELCOR severe accident scenarios for all chemical groups, as can be observed in Table 1 of this RAI response (which will replace Table 5-9 of TR-0915-17565-P, Rev. 2 in the forthcoming TR-0915-17565-P, Rev. 3, with the exceptions that the “mean” column of Table 1 of this RAI response will not be included and the RG 1.183 and SAND 2011-0128 columns will be retained in the equivalent table in TR-0915-17565-P, Rev. 3).

The suggestion of evaluating the offsite radiological consequences for each of the five scenarios individually would imply a search for a single worst case event. However, RG 1.183 regulatory position 2.3 states “The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events.” NuScale sought to represent a spectrum of severe accidents by using a median release fraction, the minimum onset time for fission product release from the gap, and the minimum duration of the release determined from the spectrum of source term design basis accidents (STDBAs) to evaluate the offsite radiological consequences.

2. {{

}}2(a),(c)

{{

}}2(a),(c)

3. An inert species ratio range of 2-4 was investigated in Section 4.4.2 of TR-0915-17565-P, Rev. 2 and showed that a smaller value of inert ratio produces more conservative dose results (as expected). Only an inert species ratio of 2 is utilized in NuScale containment aerosol removal analysis associated with the core damage MHA. An inert species ratio value of 2 is derived from core degradation experimental and theoretical data (Reference 7.2.43 of TR-0915-17565-P, Rev. 2). A survey of core damage experiments described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species mass composition ratio range of 1-3. Additionally, theoretical mass compositions based on a chemical equilibrium estimate of aerosolized core materials described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species ratio greater than 3. An inert aerosol species ratio of 2 therefore constitutes an average representative modeling surrogate for all inert materials forming aerosols based on experimental data, and a conservative modeling surrogate for all inert materials forming aerosols based on chemical equilibrium estimates of aerosolized core materials.

Although it is observed that the NuScale fuel assembly design includes M5 material, which does not include tin and for which little core degradation experimental data exists, it can be inferred that the contribution of tin-based inert aerosols to total aerosol composition is largely offset by the contribution of boron-based aerosols in the expected post-accident NuScale module steam environment. Further,tin inclusions remain in the NuScale fuel assembly structural components and therefore the generation of tin-based aerosols from a postulated NuScale core degradation is not entirely precluded.

Although Section 6 of "NuScale-HTP2TM Fuel and Control Rod Assembly Designs", TR-0813-51127-P, Rev. 1 shows that approximately {{ }}[2(a),(c)] of the total absorber material volume in the NuScale control rod assembly is composed of B_4C absorber material, for which little core degradation experimental data exists, it is observed that the lower absorber material in the control rods is the more typical silver-indium-cadmium (Ag-In-Cd) material, and the control rod cladding is 304 stainless steel tubing with stainless steel end plugs welded to each end. Therefore, some amount of typical material effects associated with available experimental aerosol generation data that would tend to raise the inert species ratio would still be expected to occur. Further, the observation that B_4C absorber material could oxidize to form aerosols is offered by multiple sources (TR-0915-17565-P, Rev. 2 References 7.2.10 and 7.2.43 and Reference 1 of this RAI response). The theoretical contribution of boron control material to overall aerosol composition of boiling water reactors (BWRs) is estimated to be as high as 75.1% per page 387 of Reference 2 of this RAI response. It is noted that known experimental information for the contribution of boron control material to aerosol composition is limited to core degradation experiment FPT3 (Reference 1 of this RAI response). FPT3 involved a steam-poor environment resulting in the limited oxidation of boron control material and, in turn, an overweighting of the contribution to vaporized core fraction of structural elements such as tin compared to the expected steam environment of a NuScale STDBA.

It is additionally noted, as already discussed in Enclosure 1 to NuScale letter to NRC dated February 6, 2017 LO-0117-52870 (ML17037D391), that the NuScale design inherently contains a higher ratio of structural materials to radioactive materials than conventional pressurized water reactors and BWRs. Although the contribution of structural materials to total aerosol may be less significant than the contribution of control materials, the higher NuScale structural material proportion nonetheless constitutes an aspect of the NuScale inert species ratio which would be greater than typical.

Finally, it is noted that the primary benefit of inert species abundance from an aerosol deposition standpoint is that it increases the suspended particle concentration without increasing the radioactive material. This has the benefit of increasing the coagulation rate and therefore the sedimentation rate (due to particle size). This has less of an impact on diffusiophoresis and thermophoresis, which will be credited in TR-0915-17565-P, Rev. 3, which are more dependent on system conditions than on particle concentration. These phenomena have a greater impact on the NuScale design as the relatively cold pool that touches the containment can drive these processes more strongly than the atmospheric boundary of a traditional containment. Therefore, while a lower inert species ratio is more conservative, it is expected to be of less significance in aerosol removal for the NuScale design than for a traditional light water reactor.

Therefore, based on the preceding observations, NuScale's utilization of an inert species ratio value of 2 is a reasonable assumption.

4. Figure 5-13 of TR-0915-17565-P, Rev. 2 was in error because of an internal post-processing error in the STARNAUA software. The vendor has been notified and the error has been addressed. NuScale will update Figure 5-13 and other associated STARNAUA example results in TR-0915-17565-P, Rev. 3. Figure 2 of this RAI response shows Figure 5-13's replacement that will be incorporated into TR-0915-17565-P, Rev. 3.

Description		STDBA No. 1	STDBA No. 2	STDBA No. 3	STDBA No. 4	STDBA No. 5	Median	Mean
onset of gap release (hr)		17.6	3.8	8.1	6.2	21.3	8.1	11.4
duration of gap plus early in-vessel release (hr)		12.0	1.0	9.0	1.3	14.0	9.0	7.5
fraction of initial core inventory released into containment	noble gases	0.39	0.19	0.41	0.19	0.48	0.39	0.33
	halogens	0.21	3.5E-02	0.16	1.9E-02	0.14	0.14	0.11
	alkali metals	0.25	5.9E-02	0.22	3.1E-02	0.20	0.20	0.15
	alkaline earths	5.9E-03	2.8E-03	6.7E-03	2.4E-03	5.3E-03	5.3E-03	4.6E-03
	tellurium group	0.22	3.8E-02	0.16	2.3E-02	0.15	0.15	0.12
	molybdenum	6.4E-02	1.3E-02	5.3E-02	5.8E-03	4.9E-02	4.9E-02	3.7E-02
	noble metals	1.2E-03	1.2E-04	1.5E-03	4.9E-05	7.9E-04	7.9E-04	7.3E-04
	lanthanides	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08
	cerium group	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08

Table 1: Comparison of release timing and magnitudes of example STDBAs

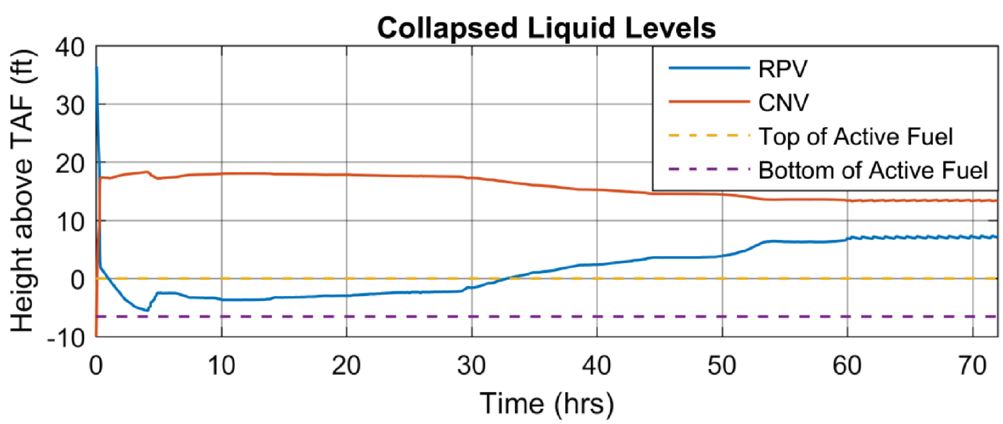

Figure 1: Example STDBA No. 2 reactor pressure vessel (RPV) and CNV collapsed liquid levels

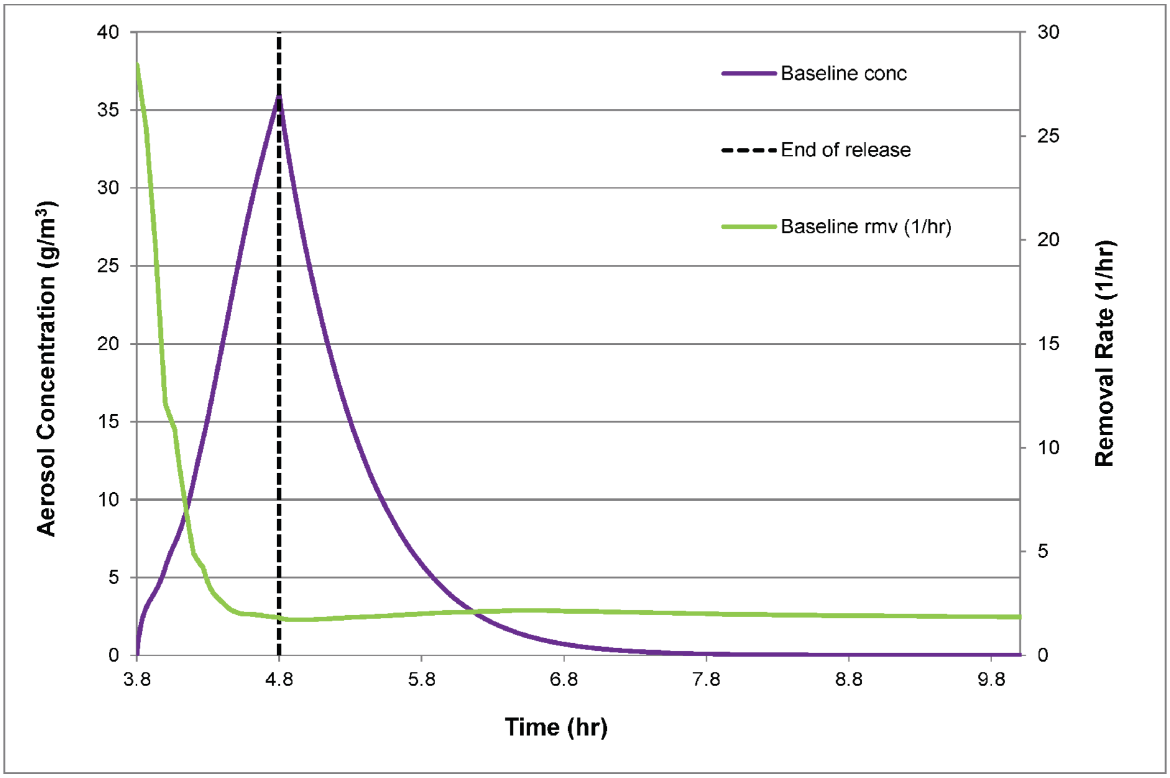

Figure 2: Baseline case aerosol concentration and removal

Reference 1: T. Haste, et al., "Phébus FPT3: Overview of Main Results concerning Core Degradation and Fission Product Behaviour", Paper 11147, Proceedings of ICAPP 2011, Nice, France, May 2011.

Reference 2: Robert P. Wichner & Roger D. Spence (1985) “A Chemical Equilibrium Estimate of the Aerosols Produced in an Overheated Light Water Reactor Core”, Nuclear Technology, 70:3, 376-393, DOI: 10.13182/NT85-A15964.

Additional Information:
TR-0915-17565-P will be revised as described in the response above.

February 28, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9646 (eRAI No. 9646) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9646 (eRAI No. 9646)," dated February 18, 2019
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2, dated April 2016

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Question from NRC eRAI No. 9646:

- 01.05-33

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9646 (eRAI No. 9646). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Paul Infanger at 541-452-7351 or at pinfanger@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

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Samuel Lee, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9646, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9646, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0219-64725

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9646, nonproprietary

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Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9646
Date of RAI Issue: 02/18/2019

NRC Question No.: 01.05-33

Regulatory basis

10 CFR 52.47(a)(2) requires, in part, a description and analysis of engineered safety features and barriers that must be breached before a release of radioactive material to the environment can occur. In performing this assessment, an applicant shall assume a fission product release from the core into the containment and use the expected demonstrable containment leak rate. 10 CFR 52.47(a)(27) requires a description of the design-specific probabilistic risk assessment and its results. 10 CFR 51.55 requires an environmental report addressing the costs and benefits of severe accident mitigation design alternatives (SAMDA).

Request for additional information

The NuScale Final Safety Analysis Report (FSAR) Revision 2 uses the methodology in the NuScale Accident Source Term Methodology topical report TR-0915-17565-P, Revision 2, to calculate radiological consequences. Section 3.3.7 of the topical report states that the containment is assumed to leak at the design basis limit leak rate for 24 hours and then at half of the design basis limit leak rate thereafter. Table 12.2-28 of the FSAR implements this assumption as 0.2% per day for the first 24 hours of the accident and 0.1% per day after 24 hours. Section 6.2.6 of the FSAR states that the specified maximum allowable containment leak rate, La, is 0.20 weight percent of the containment air mass per day at the calculated peak accident pressure, Pa, identified in Section 6.2.1.

In discussions with NRC staff, NuScale stated that the topical report and the FSAR implement the containment leak rate assumptions in Regulatory Guide 1.183. However, the containment leak rate assumptions in Regulatory Guide 1.183 are based on containment designs which have a larger containment air mass compared to NuScale's evacuated containment design. This difference is illustrated by a staff independent MELCOR confirmatory calculation for a NuScale severe accident scenario using a containment hole sized to give a containment leak rate of 0.2% per day when the containment is filled with air at 1000 psia and 72 F. The staff's calculation predicted a leak rate of 0.7% per day following core damage and that the 0.7% percent per day leak rate would continue beyond 24 hours. The higher leak rate is due to the higher severe accident mole fractions of hydrogen and steam (which are less dense than air) in containment for the NuScale design. The leak rate is scenario-dependent because the amount of hydrogen generated is scenario-dependent. The leak rate also could depend on the amount of xenon and krypton released.

Question

The staff has determined that a containment leak rate of 0.7% per day could result in a larger release of radioactive material to the environment and higher radiological consequences. As such, NuScale is requested to provide technical justification in the topical report for the containment leak rate assumed in the MHA radiological consequence assessment, including the reduction in the leak rate at 24 hours; or to revise the topical report to use a containment leak rate applicable to NuScale accident scenarios. The technical justification should address how the basis for technical specification containment leakage rate requirements is reflected in the assumed containment leakage rate during an accident. If the containment leak rate is changed in the topical report, NuScale should provide revisions to documents that are affected by this change, including the assumptions and results in FSAR 15.0.3, "Design Basis Accident Radiological Consequence Analysis for Advanced Light Water Reactors," in FSAR section 19.1.4.2.1.4, "Release Categories," in the Environmental Report, and in the EPZ Topical Report.

NuScale Response:

NuScale has followed industry precedents in utilizing RG 1.183. Additionally, NuScale has independently verified the applicability of the RG 1.183 leak rate assumptions to NuScale, as will be described in this response.

The containment leak rate assumptions of RG 1.183 remain valid for the NuScale design, despite the smaller volume and air mass of the NuScale containment (CNV) compared to the larger containments contemplated by RG 1.183. The NuScale CNV design is similar to Mark I and Mark II boiling water reactor (BWR) containments with respect to how hydrogen and steam would affect the leak rate. NuScale severe accident simulations result in approximately 30 to 100 percent metal-water reaction of fuel cladding, which correspond to maximum hydrogen concentration values of approximately 82 to 97 percent by volume. A 30 to 100 percent metal-water reaction of the cladding in a postulated loss of coolant accident (LOCA) at a BWR (Mark I and Mark II containments) is estimated to result in a containment atmosphere with a hydrogen concentration by volume of approximately 44 to 73 percent (U.S. Nuclear Regulatory Commission, "Light Water Reactor Hydrogen Manual," NUREG/CR-2726 SAND82-1137, Rev. 3, August 1983). In this respect, there is precedent for light water reactor (LWR) containments with the potential to become hydrogen-rich during a severe accident scenario to assume RG 1.183 leak rate assumptions based on containment air mass leakage rather than hydrogen or steam based leakage rates.

There is no industry precedent for specifying containment leakage based on anything besides an air atmosphere, and existing regulatory guidance (U.S. Nuclear Regulatory Commission, "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," NUREG/CR-6604, Rev. 0, April 1998) recommends the use of RADTRAD, which employs an air leakage transport mechanism for containment leakage modeling. This practice is standard even though it is known that core damage accidents in operating reactor designs have resulted in accident gas atmospheres containing hydrogen. The assumption of dry air technical specification containment leakage in accident dose evaluations is also standard industry practice, even though RG 1.183 requires the evaluation of a LOCA with core damage and the core damage would result in the presence of hydrogen in the containment atmospheres.

Although the staff does not describe the underlying assumptions or methodology choices of their independent MELCOR confirmatory calculation in RAI 9646, NuScale has independently performed leak rate calculations by multiple methods, including an orifice flow leak rate estimation resulting in values similar to those provided by the staff. Additionally, during an October 3, 2018 public teleconference with the staff on their earlier, preliminary independent analysis, the orifice flow assumption was stated as being used. It is therefore inferred that the staff independent leak rate calculation applied an orifice flow assumption.

NuScale believes an orifice flow assumption is not a reasonable assumption for estimating a leak rate by which conservatism of the NuScale design basis leak rate should be comparatively judged. Per Battelle Pacific Northwest Laboratories report "Estimation of Gas Leak Rates Through Very Small Orifices and Channels", BNWL-2223, February 1977, it is recognized that with respect to accident leak rate estimation the "only definitive assertion, which can be made, is a statement on the maximum possible leak rate, which would result if the leak were assumed to be an orifice." NuScale does not believe estimation of the maximum possible leak rate in this fashion constitutes a reasonable estimate for judgment of conservatism of the NuScale design basis leak rate.

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The technical specification leakage value is an arbitrarily selected analytical limit determined to result in acceptable radiological consequences when applied as an air leakage value in RADTRAD dose analysis. Because the safety analysis analytical limit (or technical specification) CNV leakage value is an arbitrary value completely unassociated with any physically accurate estimate of leak area for a severe accident, it stands that an accident leak rate derived from the safety analysis analytical limit is also an arbitrary value.

{{

}}2(a),(c) Therefore, the containment leak rate assumptions of RG 1.183 remain valid for the NuScale design.

NuScale is not updating the topical report to include the justifications provided in this RAI response.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17565, Accident Source Term Methodology, as a result of this response.

April 19, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
Rockville, MD 20852-2738

SUBJECT: NuScale Power, LLC Supplemental Response to NRC Request for Additional Information No. 9224 (eRAI No. 9224) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9224 (eRAI No. 9224)," dated December 04, 2017
2. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9224 (eRAI No.9224)," dated April 17, 2019
3. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 2, dated April 2016
4. NuScale Topical Report, "Accidental Source Term Methodology," TR-0915-17565, Revision 3, dated April 2019

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) supplemental response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's supplemental response to the following RAI Question from NRC eRAI No. 9224:

- 01.05-32

Enclosure 1 is the proprietary version of the NuScale Supplemental Response to NRC RAI No. 9224 (eRAI No. 9224). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

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If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9224, proprietary

Enclosure 2: NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0419-65260

Enclosure 2:

NuScale Supplemental Response to NRC Request for Additional Information eRAI No. 9224, nonproprietary

NUSCALE
Power for all humankind

Response to Request for Additional Information

eRAI No.: 9224
Date of RAI Issue: 12/04/2017

NRC Question No.: 01.05-32

Regulatory basis

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are done, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; 10 CFR 50, Appendix A, GDC 19 for control room radiological habitability; and the requirements related to the technical support center in 10 CFR 50.47(b)(8) and (b)(11) and Paragraph IV.E.8 of Appendix E to 10 CFR Part 50. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Request for additional information

In the NRC staff's review of the topical report (TR), "Accident Source Term Methodology," TR-0915- 17565-P, Rev. 2, the staff requires the following information to complete its review. Also, the staff requests that the requested information be included in the TR, as appropriate.

1. The proposed methodology for determining the design basis source term in Section 4.2.1 includes using the median release fraction from the MELCOR calculations for the more likely severe accident scenarios. The NuScale Final Safety Analysis Report (FSAR) implements the methodology by taking the median over four MELCOR calculations. What is the basis for using the median release fraction as opposed to another statistical metric such as a mean or a 75th percentile? Given the limited number of scenarios (four) as implemented in the FSAR, what is the basis for using a median release fraction to evaluate

the offsite radiological consequences as opposed to evaluating the offsite radiological consequences for each of the four scenarios?

2. The second paragraph of Section 4.2.4 describes the methodology {{ }}[2(a),(c)] {{ }}[2(a),(c)] However, the methodology is not clear to the staff. The applicant is requested to clarify {{ }}[2(a),(c)]

3. Enclosure 1 to NuScale letter to NRC dated February 6, 2017, provided clarification of the basis of its use of an inert species ratio (i.e., ratio of non-radioactive aerosol to radioactive aerosol) of 2 to 4. As part of the clarification, NuScale stated that past studies showed an inert species ratio of 2 to 4 and that the NuScale fuel design is based on a conventional PWR fuel design with similar geometries, materials, and fuel composition with a similar ratio of control material to fission products. The staff's subsequent review of additional NuScale documents as part of the design certification review suggests that the inert species ratio could be different from that seen in the past studies because NuScale's core is different from previous PWR designs in two ways – it does not use 1% tin in its cladding and it mainly uses B4C control rods. The applicant is requested to provide additional information supporting the use of a ratio of 2 to 4 given these two differences.

4. With respect to the example implementation of the methodology on aerosol removal, the two curves (aerosol concentration and aerosol removal rate constant) in Figure 5-13 of the TR appear to be inconsistent with each other. For the period beginning at the end of the release, the decrease in the aerosol concentration has a lower removal rate than shown in the removal rate constant curve. Please clarify this apparent inconsistency and how the curves were calculated.

NuScale Response:

1. NuScale now uses the term "core damage source term (CDST)" instead of the term "design basis source term". NuScale now utilizes the term “surrogate accident scenario” instead of “source term design basis accident”.

It should be noted that the number of surrogate accident scenarios implemented in determining the CDST had been updated from four to five, as detailed in the previously submitted May 24, 2017 response to RAI 8774, Question 15.00.03-1 (ML17144A451). The example CDST analysis shown in TR-0915-17565-P, Rev. 3 has been updated to use a set of five surrogate accident scenarios.

As noted in Section 4.2.3 of TR-0915-17565-P, Rev. 2, the use of the median is similar to the approach used in Sandia National Laboratory report SAND2011-0128 (Reference 7.2.10 of TR-0915-17565-P, Rev. 2). SAND2011-0128 gives the following justification for use of the median release fraction:

“The median is taken to be the representative value of the source term distribution… A percentile other than the median as the representative magnitude would require justification from some other source. By adopting the median, half the accidents have larger release fractions and longer release times than the representative accident and half have smaller release fractions and shorter release times.”

While SAND2011-0128 notes as a limitation that "results are not applicable to small modular reactors that could have accident processes that differ substantially from those of the large power plants considered here", this limitation applies to results rather than evaluation methods. It is NuScale’s view that the SAND2011-0128 release fraction selection method represents an applicable precedent of a solution approach to the problem of selecting representative release fractions from multiple postulated severe accident scenarios.

It is further noted that the implemented median release fractions from the MELCOR surrogate accident scenarios in the example CDST analysis are conservatively higher than the mean release fractions from the MELCOR surrogate accident scenarios for all chemical groups, as can be observed in Table 1 of this RAI response (which has replaced Table 5-9 of TR-0915-17565-P, Rev. 2 in TR-0915-17565-P, Rev. 3, with the exceptions that the “mean” column of Table 1 of this RAI response is not included and the RG 1.183 and SAND 2011-0128 columns are retained in the equivalent table (Table 5-8) in TR-0915-17565-P, Rev. 3).

The suggestion of evaluating the offsite radiological consequences for each of the five surrogate accident scenarios individually would imply a search for a single worst case event. However, RG 1.183 regulatory position 2.3 states “The AST must not be based upon a single accident scenario but instead must represent a spectrum of credible severe accident events.” NuScale sought to represent a spectrum of severe accidents by using a median release fraction and the release timing values associated with the surrogate accident scenario with the minimum time to core damage from the spectrum of surrogate accident scenarios to evaluate the offsite radiological consequences.

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3. An inert species ratio range of 2-4 was investigated in Section 4.4.2 of TR-0915-17565-P, Rev. 2 and showed that a smaller value of inert ratio produces more conservative dose results (as expected). Only an inert species ratio of 2 is utilized in NuScale containment aerosol removal analysis associated with the CDST. An inert species ratio value of 2 is derived from core degradation experimental and theoretical data (Reference 7.2.43 of TR-0915-17565-P, Rev. 2). A survey of core damage experiments described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species mass composition ratio range of 1-3. Additionally, theoretical mass compositions based on a chemical equilibrium estimate of aerosolized core materials described in Reference 7.2.43 of TR-0915-17565-P, Rev. 2 yields an inert-to-radioactive species ratio greater than 3. An inert aerosol species ratio of 2 therefore constitutes an average representative modeling surrogate for all inert materials forming aerosols based on experimental data, and a conservative modeling surrogate for all inert materials forming aerosols based on chemical equilibrium estimates of aerosolized core materials.

Although it is observed that the NuScale fuel assembly design includes M5 material, which does not include tin and for which little core degradation experimental data exists, it can be inferred that the contribution of tin-based inert aerosols to total aerosol composition is largely offset by the contribution of boron-based aerosols in the expected post-accident NuScale module steam environment. Further, tin inclusions remain in the NuScale fuel assembly structural components and therefore the generation of tin-based aerosols from a postulated NuScale core degradation is not entirely precluded.

Although Section 6 of "NuScale-HTP2TM Fuel and Control Rod Assembly Designs", TR-0813-51127-P, Rev. 1 shows that approximately {{ }}[2(a),(c)] of the total absorber material volume in the NuScale control rod assembly is composed of B_4C absorber material, for which little core degradation experimental data exists, it is observed that the lower absorber material in the control rods is the more typical silver-indium-cadmium (Ag-In-Cd) material, and the control rod cladding is 304 stainless steel tubing with stainless steel end plugs welded to each end. Therefore, some amount of typical material effects associated with available experimental aerosol generation data that would tend to raise the inert species ratio would still be expected to occur. Further, the observation that B_4C absorber material could oxidize to form aerosols is offered by multiple sources (TR-0915-17565-P, Rev. 2 References 7.2.10 and 7.2.43 and Reference 1 of this RAI response). The theoretical contribution of boron control material to overall aerosol composition of boiling water reactors (BWRs) is estimated to be as high as 75.1% per page 387 of Reference 2 of this RAI response. It is noted that known experimental information for the contribution of boron control material to aerosol composition is limited to core degradation experiment FPT3 (Reference 1 of this RAI response). FPT3 involved a steam-poor environment resulting in the limited oxidation of boron control material and, in turn, an overweighting of the contribution to vaporized core fraction of structural elements such as tin compared to the expected steam environment of a NuScale surrogate accident scenario.

It is additionally noted, as already discussed in Enclosure 1 to NuScale letter to NRC dated February 6, 2017 LO-0117-52870 (ML17037D391), that the NuScale design inherently contains a higher ratio of structural materials to radioactive materials than conventional pressurized water reactors and BWRs. Although the contribution of structural materials to total aerosol may be less significant than the contribution of control materials, the higher NuScale structural material proportion nonetheless constitutes an aspect of the NuScale inert species ratio which would be greater than typical.

Finally, it is noted that the primary benefit of inert species abundance from an aerosol deposition standpoint is that it increases the suspended particle concentration without increasing the radioactive material. This has the benefit of increasing the coagulation rate and therefore the sedimentation rate (due to particle size). This has less of an impact on diffusiophoresis and thermophoresis, which is credited in TR-0915-17565-P, Rev. 3, which are more dependent on system conditions than on particle concentration. These phenomena have a greater impact on the NuScale design as the relatively cold pool that touches the containment can drive these processes more strongly than the atmospheric boundary of a traditional containment. Therefore, while a lower inert species ratio is more conservative, it is expected to be of less significance in aerosol removal for the NuScale design than for a traditional light water reactor.

Therefore, based on the preceding observations, NuScale's utilization of an inert species ratio value of 2 is a reasonable assumption.

4. Figure 5-13 of TR-0915-17565-P, Rev. 2 was in error because of an internal post-processing error in the STARNAUA software. The vendor has been notified and the error has been addressed. NuScale has updated Figure 5-13 and other associated STARNAUA example results in TR-0915-17565-P, Rev. 3. Figure 2 of this RAI response shows Figure 5-13's replacement that has been incorporated into TR-0915-17565-P, Rev. 3 as Figure 5-13.

Description		Case 1	Case 2	Case 3	Case 4	Case 5	Median	Mean
onset of gap release (hr)		17.6	3.8	8.1	6.2	21.3	8.1	11.4
duration of gap plus early in-vessel release (hr)		12.0	1.0	9.0	1.3	14.0	9.0	7.5
fraction of initial core inventory released into containment	noble gases	0.39	0.19	0.41	0.19	0.48	0.39	0.33
	halogens	0.21	3.5E-02	0.16	1.9E-02	0.14	0.14	0.11
	alkali metals	0.25	5.9E-02	0.22	3.1E-02	0.20	0.20	0.15
	alkaline earths	5.9E-03	2.8E-03	6.7E-03	2.4E-03	5.3E-03	5.3E-03	4.6E-03
	tellurium group	0.22	3.8E-02	0.16	2.3E-02	0.15	0.15	0.12
	molybdenum	6.4E-02	1.3E-02	5.3E-02	5.8E-03	4.9E-02	4.9E-02	3.7E-02
	noble metals	1.2E-03	1.2E-04	1.5E-03	4.9E-05	7.9E-04	7.9E-04	7.3E-04
	lanthanides	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08
	cerium group	3.3E-08	2.6E-09	3.1E-08	1.1E-09	2.1E-08	2.1E-08	1.8E-08

Table 1: Comparison of release timing and magnitudes of example surrogate accident scenario cases

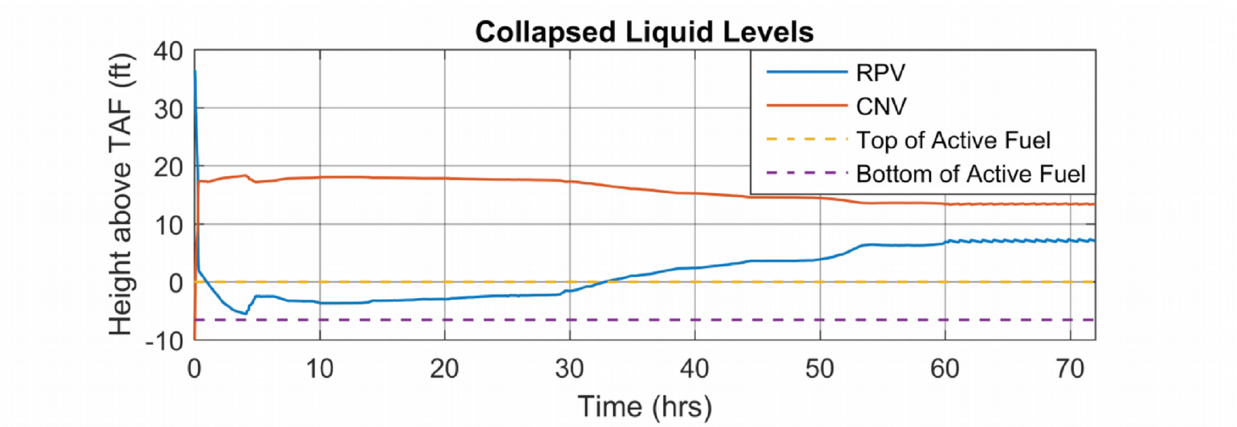

Figure 1: Example surrogate accident scenario case 2 reactor pressure vessel (RPV) and CNV collapsed liquid levels

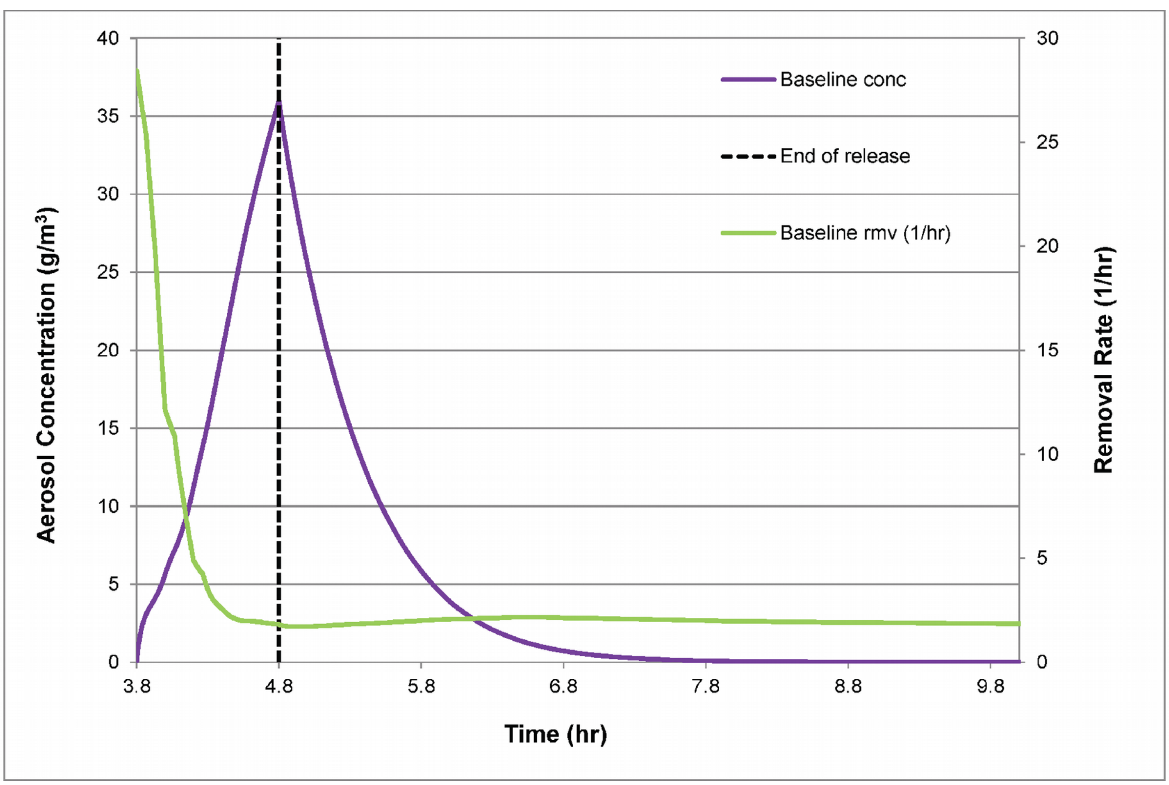

Figure 2: Baseline case aerosol concentration and removal

Reference 1: T. Haste, et al., "Phébus FPT3: Overview of Main Results concerning Core Degradation and Fission Product Behaviour", Paper 11147, Proceedings of ICAPP 2011, Nice, France, May 2011.

Reference 2: Robert P. Wichner & Roger D. Spence (1985) “A Chemical Equilibrium Estimate of the Aerosols Produced in an Overheated Light Water Reactor Core”, Nuclear Technology, 70:3, 376-393, DOI: 10.13182/NT85-A15964.

Additional Information:

TR-0915-17565-P Revision 3 is consistent with the descriptions provided in the response above.

July 31, 2019

Docket: PROJ0769

U.S. Nuclear Regulatory Commission
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9690 (eRAI No. 9690) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9690 (eRAI No. 9690)," dated June 27, 2019
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3, dated April 2019

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosures to this letter contain NuScale's response to the following RAI Questions from NRC eRAI No. 9690:

- 01.05-39
- 01.05-41
- 01.05-42

Enclosure 1 is the proprietary version of the NuScale Response to NRC RAI No. 9690 (eRAI No. 9690). NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 is the nonproprietary version of the NuScale response.

This letter and the enclosed responses make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
Getachew Tesfaye, NRC, OWFN-8H12

Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9690, proprietary

Enclosure 2: NuScale Response to NRC Request for Additional Information eRAI No. 9690, nonproprietary

Enclosure 3: Affidavit of Zackary W. Rad, AF-0719-66518

Enclosure 2:

NuScale Response to NRC Request for Additional Information eRAI No. 9690, nonproprietary

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9690

Date of RAI Issue: 06/27/2019

NRC Question No.: 01.05-39

Regulatory Basis:

10 CFR 50.49(e)(4) requires that the radiation environment for equipment qualification must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

Background:

On April 21, 2019, NuScale submitted Revision 3 to TR-0915-17565, "Licensing Topical Report Accident Source Term Methodology." The revision included a new design basis iodine spike source term and reclassified the maximum hypothetical accident as a beyond design basis source term (DBST). This resulted in the maximum hypothetical accident no longer being considered for environmental qualification and the iodine spike source term being used for the maximum radiation environment being used for equipment qualification in and around containment.

In TR-0915-17565, Revision 3, Section 3.2.6, the applicant indicates that, "Spiking effects may occur for radionuclides besides iodines. However, any potential spiking of radionuclides besides iodine is implicitly accounted for by conservative treatments of the iodine spike DBST. For example, the assumed instantaneous event time-zero release of the entire primary coolant inventory results in doses expected to be several times larger than a more realistic graduated release of a primary coolant mass less than the entire primary coolant mass." The staff understands that assuming an instantaneous release may be conservative, but TR-0915-17565 does not provide information explaining NuScale's statement that the conservatisms bound the consideration of spiking of other radionuclides.

The applicant also does not provide any additional information or justification of the implicit conservatism to support their position except that the treatment of primary coolant activity, including iodine spiking, is consistent with RG 1.183. However, RG 1.183 assumes that a core melt accident is being considered for the radiation environment for equipment qualification, which typically bounds the dose to equipment inside containment. Since a core melt source term is not being considered for NuScale, additional justification is needed for why it is not necessary to consider the spiking of other radionuclides besides iodine for equipment impacted by the iodine spike design basis source term.

Issue:

Additional information is needed to demonstrate the conservatisms in developing the iodine spike DBST, as the staff is unable to make a determination that the radiation environment associated with the most severe design basis accident is being appropriately considered for environmental qualification.

Request:

Please provide 1) justification that the methodology used for developing the design basis iodine spike reactor coolant source term, described in TR-0915-17565 provides a source term that reasonably conservatively bounds the radiation environment associated with the most severe design basis accident, as required by 10 CFR 50.49(e)(4) or 2) update the topical report, as appropriate, to ensure that the methodology appropriately considers the potential for spiking of other radionuclides besides iodine or bounds the potential spiking of other radionuclides.

NuScale Response:

TR-0915-17565 has been revised (see specifically Appendix B and Section B.3.1.1) to provide further justification that the methodology used for developing the design basis iodine spike reactor coolant source term provides a source term that conservatively bounds the radiation environment associated with the most severe design basis accident, as required by 10 CFR 50.49(e)(4).

Impact on Topical Report:

Topical Report TR-0915-17565, Accident Source Term Methodology, has been revised as described in the response above and as shown in the markup provided in this response.

Appendix B. Environmental Qualification Dose Analysis Methodology

This appendix describes the methodology for calculating environmental qualification (EQ) doses in the CNV and bioshield envelope regions. The methodology is for immersion dose rates, photon shine, total integrated radiation doses, and energy deposited for the specified CNV and bioshield envelope regions. {{

}}2(a),(c)

B.1 EQ Dose Methodology Evaluation Scenarios

The goal of this EQ dose methodology is to identify and evaluate a conservative surrogate for the worst-case design basis accident (DBA) for radiation exposures to equipment in the CNV and in the bioshield envelope. The conservative surrogate for the worst-case DBA is identified for each region in the following fashion:

- For equipment in the lower CNV (sump) liquid region – {{

  }}2(a),(c) (Section B.1.1.1)

- For equipment in the upper CNV vapor region – {{

  }}2(a),(c)

  (Section B.1.1.2)

- For equipment in the bioshield envelope – {{

  }}2(a),(c) (Section B.1.2)

Further details of the conservative nature of this EQ dose methodology is provided in the following sections.

B.1.1 Containment Release General Scenario

The nature of a direct primary coolant (plus iodine spike) release to the CNV, as applied in CNV EQ dose evaluations, is conservative. {{

}}2(a),(c)

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}}2(a),(c) This treatment of the iodine spike release timing is conservative.

The containment analysis is performed for two separate regions (the upper CNV vapor region and the lower CNV liquid region). {{

}}2(a),(c) This method of defining the CNV regions for either containment

analysis scenario conservatively confines total source inventory to a smaller volume than that of the total CNV free volume.

B.1.1.1 Lower Containment Liquid Region Evaluation Scenario

For the purposes of evaluating the dose to equipment in the lower CNV liquid region, {{

}}2(a),(c)

B.1.1.2 Upper Containment Vapor Region Evaluation Scenario

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}}2(a),(c)

B.1.2 Bioshield Envelope Evaluation Scenario

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}}2(a),(c)

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}}2(a),(c)

The bioshield envelope evaluation scenario described above is conservative {{

}}2(a),(c)

B.2 Assumptions

B.2.1 Activity Plated Out on Containment Surfaces

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}}2(a),(c)

B.2.2 Activity Release Timing

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}}2(a),(c)

B.2.3 Liquid and Vapor RCS Densities

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}}2(a),(c)

B.2.4 Credit for Natural Mechanisms

As stated in Regulatory Guide 1.183, Appendix A, credit may be taken for reduction in the available amount of radiation due to natural deposition mechanisms. {{

}}2(a),(c)

B.2.5 CVCS Purification for Coincident Iodine Spike Calculation

The primary coolant iodine concentration is estimated using a "spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value" (Regulatory Guide 1.183, Appendix E, Item 2.2). {{

}}2(a),(c)

B.2.6 Decay Chain

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}}2(a),(c)

B.2.7 Medium Model

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}}2(a),(c)

B.2.8 Time-Dependent Containment Leak Rate

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}}2(a),(c)

B.3 Methodology

B.3.1 Primary Coolant Source Term

For the EQ dose evaluation, the primary coolant radionuclide inventory described in Section 3.3.2 of this report, including isotopic concentrations equivalent to the design basis DE I-131 and DE Xe-133 limits, is applied.

B.3.1.1 Non-Iodine Spiking

Spiking of radionuclides besides iodine is not explicitly considered in this methodology. This approach is consistent with the available regulatory guidance, which does not prescribe the spiking of radionuclides besides iodines. Regulatory Guide 1.183, Appendix I, addresses assumptions for evaluating radiation doses for equipment qualification purposes. Regulatory Position 4 therein notes the possibility that “another design basis accident” (i.e., non-core melt events) may be more limiting than the "design basis LOCA" (i.e., the core melt source term event) for the purposes of equipment qualification for some components. In these cases, RG 1.183 recommends the use of the applicable assumptions of Appendices B through H otherwise applicable to the dose consequence evaluations for the event in question, which do not include spiking of any radionuclides other than iodines. A key use of the iodine spike DBST is to establish the radiation environment for a design basis accident inside containment, as other events are expected to be more limiting with respect to dose consequences outside containment. Therefore, NuScale concludes the existing guidance of RG 1.183, which establishes the assumptions of Appendices B through H are adequate for a similar use (developing design-basis EQ doses), is appropriate to follow with respect to the iodine spike DBST. {{

}}2(a),(c)

B.3.1.2 Coincident Iodine Spiking

The coincident iodine spike modeling approach used in this methodology {{

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B.3.2 Energy, Dose Rates, and Integrated Dose

The total energy rate for a given isotope is based upon its initial activity and average energy per decay. To calculate the activity of an isotope sometime after shutdown, a standard exponential decay model is used to extrapolate the values based on isotopic half-lives, as described by Eq. B-1.

$$A_i(t) = A_{o,i} e^{\frac{-ln|2|t}{T_{1/2}}}$$ Eq. B-1

where,

$A_i(t)$ = Activity of isotope i at time t, Ci

$A_{o,i}$ = Initial activity of isotope i, Ci

$T_{1/2}$ = Half-life for isotope =i, s

t = time at which to calculate the activities, s

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}}2(a),(c) With activities determined for a given hourly interval, photon or electron energy emission rate in units of MeV/s are calculated based on the average photon or electron emission rate for a single disintegration, or nuclear transformation. The unit “nt”, an abbreviation for nuclear transformation, is used. This “nt” unit is equal to one becquerel (Bq). Multiplying an activity, “A”, by the average energy, “E”, results in the energy emission rate, “R”, given by

$$[A]Ci \cdot \frac{3.7E10 \frac{nt}{s}}{Ci} \cdot [E]\frac{MeV}{nt} \rightarrow [R]\frac{MeV}{s}$$ Eq. B-3

Finally, the dose rate in units of rad/hr may be calculated based on the energy emission rate, volume and density of interest, and several unit conversions as expressed by

$$[R]\frac{MeV}{s}\cdot\frac{3600\,s}{hr}\cdot\frac{1.602E-19J}{eV}\cdot\frac{10^6}{M}\cdot\frac{Rad}{0.01\frac{J}{kg}}\cdot\frac{1}{[V]\,ft^3}\cdot\frac{ft^3}{[\rho]\,\text{lbm}}\cdot\frac{2.2046\,lbm}{kg}\rightarrow\frac{Rad}{hr} \qquad \text{Eq. B-4}$$

The total photon emission energy rate or dose rate is then the sum of all the emission or dose rates for all the isotopes considered, as is the case for the total electron rates. At each time step, the rates and integrated emitted energy or dose may be calculated. The integrated energy emitted in MeV is calculated for the example 2400 hour duration as follows

$$Energy\ Emitted\ (MeV) = \sum_{i=0}^{2400}(Energy\ Rate)_i\left(\frac{MeV}{s}\right)\cdot 3600\left(\frac{s}{hr}\right)\cdot 1\ hr \qquad \text{Eq. B-5}$$

Similarly, the integrated dose in units of rad is given by

$$Dose\ (Rad) = \sum_{i=0}^{2400}(Dose\ Rate)_i\left(\frac{Rad}{hr}\right)\cdot 1\ hr \qquad \text{Eq. B-6}$$

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B.3.3 Containment Leakage

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}}2(a),(c)

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}}2(a),(c)

B.4 Summary and Conclusions

In summary, a methodology for calculating EQ doses is described. Notable conservatisms of this methodology include:

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Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9690

Date of RAI Issue: 06/27/2019

NRC Question No.: 01.05-41

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

10 CFR 50.49(e)(4) requires that the radiation environment for equipment qualification must be based on the type of radiation, the total dose expected during normal operation over the installed life of the equipment, and the radiation environment associated with the most severe design basis accident during or following which the equipment is required to remain functional, including the radiation resulting from recirculating fluids for equipment located near the recirculating lines and including dose-rate effects.

Background:

NuScale topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the iodine spike design basis source term (iodine spike DBST), which is used to show compliance with the regulatory requirements described above.

Issue:

In order to make the finding on the acceptability of the topical report's methodology, additional information is needed for the staff to understand NuScale's implementation of the methodology and assumptions for the iodine spike DBST and how the topical report methodology is used to provide the source term information in Table 12.2- 37 and dose rate information in Table 3C-8.

Request:

1. Please provide additional details in Section 3.2.6 of the topical report on the analysis assumptions for the iodine spike DBST, including bases for the assumptions, to the same level of detail as for the other design basis events. Include details such as the following:

    - Clarify the timing of the release to containment.
        - For example, clarify whether the entire integrated activity (including total coincident iodine spike values) is assumed to be released instantaneously, or is the initial RCS activity released instantaneously at time = 0, with coincident iodine spike activity appearing over 8 hours?
    - Clarify the assumptions on mixing in the containment.
    - For example, clarify the following:
        - Is the release mixed throughout entire containment air volume?
        - What is the assumed containment air volume? Is it the same value for containment air volume used in the CDE dose analysis?
    - Additionally, please revise the text in Section 3.2.6 to clarify that the iodine spike design basis source term includes 2 iodine spiking cases.

2. Provide additional detail in FSAR Section 15.0.3.8.6 on iodine spike DBST assumptions and their bases, similar to the detail for other DBAs in FSAR Section 15.0.3.8. Solely relying on a reference to the topical report does not give the staff enough information to make a safety finding. Include such information as the following:

    - Timing of release.
    - Containment mixing assumptions.
        - For example, please clarify the following:
            - Is the release mixed throughout entire containment air volume?
            - What is the assumed containment air volume?

- Is it the same value for containment air volume used in the core damage event dose analysis?

  - Assumed mass of the primary coolant.

  - Assumptions for the two iodine spiking cases.

3. Provide additional information, including updates in the FSAR and/or topical report, as appropriate, to describe the methods, models, and assumptions used in developing the source term provided in FSAR Table 12.2-37.3.

4. Please describe the methods, models, and assumptions used for calculating the total integrated doses provided in FSAR Table 3C-8. Please ensure that the discussion includes information demonstrating why the maximum design basis accident total integrated dose values provided in FSAR Table 3C-8 represent dose rates for the most severe design basis accident, with appropriate margin, as required by 10 CFR 50.49(e)(4) and 10 CFR 50.49(e)(8). Include updates to the FSAR and/or topical report, as appropriate.

NuScale Response:

1. TR-0915-17565 Section 3.2.6 has been revised to provide additional details on the analysis assumptions for the iodine spike DBST to the same level of detail as for the other design basis events. See note in response to question 2 below for information on containment volume assumptions.

2. FSAR Section 15.0.3.8.6 has been revised to provide additional details on the analysis assumptions for the iodine spike DBST similar to the amount of detail for other DBAs in FSAR Section 15.0.3.8.

Note: {{

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}}2(a),(c)

3. The RAI 9268 (question 12.02-11S1) supplemental response provides a description of the source inputs and assumptions used for the development of the "Maximum Post-Accident Radionuclide Concentrations" in Table 12.2-34. In addition to the information provided in the supplemental response to RAI 9268, the iodine spiking factor and spiking durations provided in the revised TR-0915-17565 were used in the development of the values in Table 12.2-34 "Maximum Post-Accident Radionuclide Concentrations."

4. TR-0915-17565 has been revised to include an Appendix B, entitled “Environmental Qualification Dose Analysis Methodology”, which describes the methods, models, and assumptions used for calculating the total integrated doses provided in FSAR Table 3C-8.

Impact on DCA:

FSAR Section 15.0.3 and TR-0915-17565 have been revised as described in the response above and as shown in the markup provided with this response.

results in a time-dependent release of activity to the reactor building which is modeled for conservatism as a direct release to the environment through the break.

4. After containment isolation, primary coolant leaks through one containment isolation valve (other in-series valve is assumed to fail) at the maximum leak rate allowed by design basis limits. The leakage continues until the reactor is brought to shutdown conditions. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by any intervening structures.
5. Available primary coolant in the CVCS equipment (heat exchangers, filters, etc.) and piping flows out of one or the other side of the break. The coolant is at the maximum activity concentration allowed by design basis limits.
6. Once the reactor is completely shut down and depressurized, all releases through valve leakage stops.

The following is a summary of the explicit assumptions used from Appendix E (main steam line break) of RG 1.183.

- coincident iodine spiking factor: 500
- duration of coincident iodine spike: 8 hr
- 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

3.2.6 Iodine Spike Design Basis Source Term

The iodine spike DBST is composed of a set of key parameters, derived from the assumption of a generic failure occurring inside the CNV, which results in the release of all primary coolant from the reactor coolant system (RCS) to the CNV. The iodine spike DBST is a surrogate that bounds the radiological consequences of a spectrum of events that result in primary coolant entering an intact containment.

Primary coolant with radionuclide concentrations at the design basis limits enters the containment and 100 percent of the radionuclides within 100 percent of the primary coolant are assumed to be present in the containment. This assumption is conservative because some amount of primary coolant (at least the amount required to cover the core) would remain in the reactor pressure vessel (RPV) and, therefore, the radionuclides associated with that primary coolant would not be available in the CNV for release. Additionally, this is conservative because some amount of the radionuclides would remain in the primary coolant at the bottom of the CNV, but the analysis assumes all the radionuclides are available to leak out of the CNV as vapor. Because the iodine spike DBST is not a specific event, nor an extension of a specific event, there is no thermal-hydraulic analysis associated with the iodine spike DBST.

~~This radiological consequence analysis considers the iodine spike DBST with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. A coincident iodine spiking factor of 500 is~~

~~utilized because this is the largest coincident iodine spiking factor recommended for any event in RG 1.183. The duration of the coincident iodine spike is assumed to be eight hours. The pre-incident iodine spike considered for this analysis is derived as shown in Section 3.3.2 of this report. Aerosol removal is not credited. An iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide is arbitrarily assumed because RADTRAD requires the input, but this assumption has no impact on results. No reduction or mitigation of noble gas radionuclides released from the primary coolant is assumed.~~This radiological consequence analysis considers the iodine spike DBST with two different initial iodine concentrations, one based on a pre-incident iodine spike and the other based on a coincident iodine spike. These iodine spikes are derived as shown in Section 3.3.2 of this report. A description of the evaluated scenario is summarized as follows:

1. A generic failure is assumed to occur inside the CNV, resulting in the release of all primary coolant from the RCS to the CNV.
2. The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios.
3. Primary coolant flows into the CNV through a nonspecific release point with an instantaneous release of activity into the CNV. The release is homogenously mixed as vapor throughout the entire CNV free volume.
4. Activity is then assumed to leak into the environment at the design basis leakage rate for 24 hours, then at 50 percent of the design basis leakage rate thereafter. The activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures. Aerosol removal is not credited.
5. Once the reactor is completely shut down and depressurized, all releases through the containment to the environment stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- Coincident iodine spiking factor – 500 (because this is the largest coincident iodine spiking factor recommended for any event in RG 1.183)
- Duration of coincident iodine spike – 8 hours
- Iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide (arbitrary assumption because RADTRAD requires the input, but this assumption has no impact on results)
- Activity released from the fuel due to the pre-incident iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the CNV; activity released from the fuel due to the coincident iodine spike is assumed to mix instantaneously and homogeneously within the fuel volume, then release to the CNV over the 8 hour coincident spiking duration
- No reduction or mitigation of noble gas radionuclides released from the primary system

~~Spiking effects may occur for radionuclides besides iodines. However, any potential spiking of radionuclides besides iodine is implicitly accounted for by conservative treatments of the iodine spike DBST. For example, the assumed instantaneous event-time-zero release of the entire primary coolant inventory results in doses expected to be several times larger than a more realistic graduated release of a primary coolant mass less than the entire primary coolant mass. This approach is consistent with the available regulatory guidance, which does not prescribe the spiking of radionuclides besides iodines. Regulatory Guide 1.183, Appendix I, addresses assumptions for evaluating radiation doses for equipment qualification purposes. Regulatory Position 4 therein notes the possibility that "another design basis accident" (i.e., non-core melt events) may be more limiting than the "design basis LOCA" (i.e., the core melt source term event) for the purposes of equipment qualification for some components. In these cases, RG 1.183 recommends the use of the applicable assumptions of Appendices B through H otherwise applicable to the dose consequence evaluations for the event in question, which do not include spiking of any radionuclides besides iodines. A key use of the iodine spike DBST is to establish the radiation environment for a design basis accident inside containment, as other events are expected to be more limiting with respect to dose consequences outside containment. Therefore, NuScale concludes the existing guidance of RG 1.183, which establishes the assumptions of Appendices B through H are adequate for a similar use (developing design-basis equipment qualification radiation environments), is appropriate to follow with respect to the iodine spike DBST. Accordingly, spiking of radionuclides besides iodine is not explicitly considered in the iodine spike DBST methodology.~~

Appendix B. Environmental Qualification Dose Analysis Methodology

This appendix describes the methodology for calculating environmental qualification (EQ) doses in the CNV and bioshield envelope regions. The methodology is for immersion dose rates, photon shine, total integrated radiation doses, and energy deposited for the specified CNV and bioshield envelope regions. {{

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B.1 EQ Dose Methodology Evaluation Scenarios

The goal of this EQ dose methodology is to identify and evaluate a conservative surrogate for the worst-case design basis accident (DBA) for radiation exposures to equipment in the CNV and in the bioshield envelope. The conservative surrogate for the worst-case DBA is identified for each region in the following fashion:

- For equipment in the lower CNV (sump) liquid region – {{

  }}2(a),(c) (Section B.1.1.1)

- For equipment in the upper CNV vapor region – {{

  }}2(a),(c)

  (Section B.1.1.2)

- For equipment in the bioshield envelope – {{

  }}2(a),(c) (Section B.1.2)

Further details of the conservative nature of this EQ dose methodology is provided in the following sections.

B.1.1 Containment Release General Scenario

The nature of a direct primary coolant (plus iodine spike) release to the CNV, as applied in CNV EQ dose evaluations, is conservative. {{

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}}2(a),(c) This treatment of the iodine spike release timing is conservative.

The containment analysis is performed for two separate regions (the upper CNV vapor region and the lower CNV liquid region). {{

}}2(a),(c) This method of defining the CNV regions for either containment

analysis scenario conservatively confines total source inventory to a smaller volume than that of the total CNV free volume.

B.1.1.1 Lower Containment Liquid Region Evaluation Scenario

For the purposes of evaluating the dose to equipment in the lower CNV liquid region, {{

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B.1.1.2 Upper Containment Vapor Region Evaluation Scenario

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B.1.2 Bioshield Envelope Evaluation Scenario

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}}2(a),(c)

The bioshield envelope evaluation scenario described above is conservative {{

}}2(a),(c)

B.2 Assumptions

B.2.1 Activity Plated Out on Containment Surfaces

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}}2(a),(c)

B.2.2 Activity Release Timing

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}}2(a),(c)

B.2.3 Liquid and Vapor RCS Densities

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}}2(a),(c)

B.2.4 Credit for Natural Mechanisms

As stated in Regulatory Guide 1.183, Appendix A, credit may be taken for reduction in the available amount of radiation due to natural deposition mechanisms. {{

}}2(a),(c)

B.2.5 CVCS Purification for Coincident Iodine Spike Calculation

The primary coolant iodine concentration is estimated using a "spiking model that assumes that the iodine release rate from the fuel rods to the primary coolant increases to a value 500 times greater than the release rate corresponding to the iodine concentration at the equilibrium value" (Regulatory Guide 1.183, Appendix E, Item 2.2). {{

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B.2.6 Decay Chain

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B.2.7 Medium Model

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}}2(a),(c)

B.2.8 Time-Dependent Containment Leak Rate

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}}2(a),(c)

B.3 Methodology

B.3.1 Primary Coolant Source Term

For the EQ dose evaluation, the primary coolant radionuclide inventory described in Section 3.3.2 of this report, including isotopic concentrations equivalent to the design basis DE I-131 and DE Xe-133 limits, is applied.

B.3.1.1 Non-Iodine Spiking

Spiking of radionuclides besides iodine is not explicitly considered in this methodology. This approach is consistent with the available regulatory guidance, which does not prescribe the spiking of radionuclides besides iodines. Regulatory Guide 1.183, Appendix I, addresses assumptions for evaluating radiation doses for equipment qualification purposes. Regulatory Position 4 therein notes the possibility that “another design basis accident” (i.e., non-core melt events) may be more limiting than the "design basis LOCA" (i.e., the core melt source term event) for the purposes of equipment qualification for some components. In these cases, RG 1.183 recommends the use of the applicable assumptions of Appendices B through H otherwise applicable to the dose consequence evaluations for the event in question, which do not include spiking of any radionuclides other than iodines. A key use of the iodine spike DBST is to establish the radiation environment for a design basis accident inside containment, as other events are expected to be more limiting with respect to dose consequences outside containment. Therefore, NuScale concludes the existing guidance of RG 1.183, which establishes the assumptions of Appendices B through H are adequate for a similar use (developing design-basis EQ doses), is appropriate to follow with respect to the iodine spike DBST. {{

}}2(a),(c)

B.3.1.2 Coincident Iodine Spiking

The coincident iodine spike modeling approach used in this methodology {{

}}2(a),(c)

B.3.2 Energy, Dose Rates, and Integrated Dose

The total energy rate for a given isotope is based upon its initial activity and average energy per decay. To calculate the activity of an isotope sometime after shutdown, a standard exponential decay model is used to extrapolate the values based on isotopic half-lives, as described by Eq. B-1.

$$A_i(t) = A_{o,i} e^{\frac{-ln|2|t}{T_{1/2}}}$$ Eq. B-1

where,

$A_i(t)$ = Activity of isotope i at time t, Ci

$A_{o,i}$ = Initial activity of isotope i, Ci

$T_{1/2}$ = Half-life for isotope =i, s

t = time at which to calculate the activities, s

{{

}}2(a),(c) With activities determined for a given hourly interval, photon or electron energy emission rate in units of MeV/s are calculated based on the average photon or electron emission rate for a single disintegration, or nuclear transformation. The unit "nt", an abbreviation for nuclear transformation, is used. This "nt" unit is equal to one becquerel (Bq). Multiplying an activity, "A", by the average energy, "E", results in the energy emission rate, "R", given by

$$[A]Ci \cdot \frac{3.7E10\frac{nt}{s}}{Ci} \cdot [E]\frac{MeV}{nt} \rightarrow [R]\frac{MeV}{s}$$ Eq. B-3

Finally, the dose rate in units of rad/hr may be calculated based on the energy emission rate, volume and density of interest, and several unit conversions as expressed by

$$[R]\frac{MeV}{s}\cdot\frac{3600\,s}{hr}\cdot\frac{1.602E-19J}{eV}\cdot\frac{10^6}{M}\cdot\frac{Rad}{0.01\frac{J}{kg}}\cdot\frac{1}{[V]\,ft^3}\cdot\frac{ft^3}{[\rho]\,\text{lbm}}\cdot\frac{2.2046\,lbm}{kg}\rightarrow\frac{Rad}{hr}$$ Eq. B-4

The total photon emission energy rate or dose rate is then the sum of all the emission or dose rates for all the isotopes considered, as is the case for the total electron rates. At each time step, the rates and integrated emitted energy or dose may be calculated. The integrated energy emitted in MeV is calculated for the example 2400 hour duration as follows

$$Energy\ Emitted\ (MeV) = \sum_{i=0}^{2400}(Energy\ Rate)_i\left(\frac{MeV}{s}\right)\cdot 3600\left(\frac{s}{hr}\right)\cdot 1\ hr$$ Eq. B-5

Similarly, the integrated dose in units of rad is given by

$$Dose\ (Rad) = \sum_{i=0}^{2400}(Dose\ Rate)_i\left(\frac{Rad}{hr}\right)\cdot 1\ hr$$ Eq. B-6

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}}2(a),(c)

B.3.3 Containment Leakage

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}}2(a),(c)

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}}2(a),(c)

B.4 Summary and Conclusions

In summary, a methodology for calculating EQ doses is described. Notable conservatisms of this methodology include:

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}}2(a),(c)

~~as a surrogate to the large break LOCA typically evaluated by LWRs to meet the regulatory intent of addressing the MHA. Five source term design basis accidents derived from the Level 1 PRA were used to establish the DBST described in Section 15.0.3.3.4 in accordance with the methodology of Reference 15.0-4. Parameters associated with the DBST are presented in Table 12.2-28, Table 12.2.29, and Table 12.2-30.~~

RAI 15.00.03-1

~~To address 10 CFR 52.47 (a)(2)(iv), the DBST is assumed to occur, resulting in significant core damage. Activity is assumed to be released from the fuel over a specified time period, as described in Reference 15.0-4 and presented in Table 12.2-28, and assumed to homogeneously mix in the containment atmosphere. Removal of aerosol in the containment occurs through natural deposition phenomena. The aerosol removal methodology utilizes the code STARNAUA to determine the time-dependent airborne aerosol mass and removal rates, as described in Reference 15.0-4. Activity is released to the atmosphere from the containment at the design basis leakage rate for 24 hours, and at 50% of the limit after 24 hours.~~

Reference 15.0-4 provides the methodology for the radiological consequences of the iodine spike DBST.~~MHA, based on the guidance provided in Appendix A of RG 1.183. Assumptions used from Appendix A of RG 1.183 are:~~

- ~~The chemical form of radioiodine released to the containment atmosphere is 95% cesium iodide (CsI), 4.85% elemental iodine, and 0.15% organic iodide. Note that the methodology considers cesium iodide as an aerosol.~~
- ~~The radioactivity released from the fuel is assumed to mix instantaneously and homogeneously throughout the free air volume of the containment.~~

This radiological consequence analysis considers the iodine spike DBST 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 evaluated scenario is summarized as follows:

1) A generic failure is assumed to occur inside the CNV, resulting in the release of all 46,700 kg of primary coolant from the RCS to the CNV.

2) The iodine and noble gas coolant activity is calculated based on the maximum concentrations allowed by design basis limits for each of the iodine spiking scenarios. The primary coolant contains a concentration of 3.7E-02 μCi/gm DE I-131 for the coincident iodine spike scenario and 2.2 μCi/gm DE I-131 for the pre-incident iodine spike scenario. For both iodine spiking scenarios, the primary coolant contains 10 μCi/gm DE Xe-133.

3) Primary coolant flows into the CNV through a nonspecific release point with an instantaneous release of activity into the CNV. The release is homogenously mixed as vapor throughout the entire CNV free volume.

4) Activity is then assumed to leak into the environment at the design basis leakage rate for 24 hours, then at 50 percent of the design basis leakage rate thereafter. The

activity from this leak path is also assumed to flow directly to the environment with no mitigation or reduction by intervening structures.

5) At 30 hours, it is assumed the reactor is shut down and depressurized and releases through the containment to the environment stop.

The following is a summary of the assumptions used from Appendix E (main steam line break) of RG 1.183:

- Coincident iodine spiking factor - 500
- Duration of coincident iodine spike - 8 hours
- Iodine chemical form of 97 percent elemental iodine and 3 percent organic iodide
- Activity released from the fuel due to the pre-incident iodine spike is assumed to mix instantaneously and homogeneously within the primary coolant in the CNV; activity released from the fuel due to the coincident iodine spike is assumed to mix instantaneously and homogeneously within the fuel volume, then release to the CNV over the 8 hour coincident spiking duration
- No reduction or mitigation of noble gas radionuclides released from the primary system

RAI 15.00.03-1

~~The radioactive source term is calculated from the maximum core inventory provided in Table 11.1-2, multiplied by the release fractions provided in Table 12.2-29. The timing of the release and the radionuclide groups assumed, and the iodine removal mechanisms in the containment are provided in Table 12.2-28 and Table 12.2-30.~~

RAI 15.00.03-1

RADTRAD is used to determine the dose, as outlined in Section 15.0.3.3.8. There are no single failures assumed for this event. The control room model is described in Section 15.0.3.7.1. The potential radiological consequences of the iodine spike DBST are presented in Table 15.0-12.

15.0.4 Safe, Stabilized Condition

Safety analyses of design basis events are performed from event initiation until a safe, stabilized condition is reached. A safe, stabilized condition is reached when the initiating event is mitigated, the acceptance criteria are met and system parameters (for example inventory levels, temperatures and pressures) are trending in the favorable direction. For events that involve a reactor trip, system parameters continue changing slowly as decay and residual heat are removed and the RCS continues to cool down. No operator action is required to reach or maintain a safe, stabilized condition.

Two additional considerations are discussed to show that Chapter 15 acceptance criteria are not challenged beyond the safe, stabilized condition. Long term decay and residual heat removal is discussed in Section 15.0.5 and a potential return to power is discussed in Section 15.0.6.

Response to Request for Additional Information
Docket: PROJ0769

eRAI No.: 9690

Date of RAI Issue: 06/27/2019

NRC Question No.: 01.05-42

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Background:

NuScale topical report TR-0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the core damage event (CDE), which is used to show compliance with the regulatory requirements described above. Changes to FSAR Chapter 15 were submitted by letter dated April 19, 2019, including description of the CDE accident analysis, which implements the topical report methodology for the NuScale design certification application (DCA).

As discussed in NuScale's supplemental response to RAI 9224 dated April 9, 2019, the CDE source term methodology considers a set of five severe-accident scenarios and takes the median value for the release fraction to the containment for each radionuclide group to provide a representative (not bounding) source term. However, it is unclear how uncertainty related to the core damage and release phenomena is accounted for in the CDE source term methodology. In other words, for each of the five severe-accident scenarios, how accurately can the release to the containment be predicted? In its independent confirmatory analysis, the staff noted that for scenario LCC- 05T the staff's predictions of iodine releases from the core at 48 hours were 90% as opposed to NuScale's prediction of 55%. The staff's confirmatory analysis is described in the staff document RES/FSCB 2019-01, "Independent MELCOR Confirmatory Analysis for NuScale Small Modular Reactor," ML19114A041 (proprietary), sent to NuScale by encrypted file in May 2019. As shown in numerous past studies for LWRs, and in particular, the staff's confirmatory analysis, there is considerable uncertainty in core heatup and degradation, specifically involving late phase core melt progression. NuScale's analysis shows only partial core damage resulting in release from the fuel of less than half of the volatile radionuclides, whereas larger amounts of core degradation and subsequent radionuclide release could potentially occur, especially in an unrecovered scenario.

Issue:

The staff notes that there is limited margin between the FSAR's CDE dose results and the control room habitability 5-rem dose limit. Because calculated dose results are generally proportional to the assumed release fraction to the containment and because NuScale's volatile radionuclide release fraction from the fuel is low, additional information is needed to clarify the treatment of uncertainty in the release fraction to the containment and its subsequent effect on the FSAR's CDE source term and dose results.

Requests:

1. Please describe in the topical report the basis and justification for assuming that partial core damage involving limited release of volatile fission products is appropriately conservative for developing the CDE release fractions to the containment to show compliance with the dose requirements given in the regulatory basis.

2. Taking into consideration the uncertainty in the release fractions to the containment, would any applicable dose criteria be exceeded for the CDE?

NuScale Response:

As noted by the staff, the source term is the median release into containment from a set of five severe accidents simulated by MELCOR. These accidents were chosen because they are representative of the major contributors to NuScale's core damage frequency. Using multiple cases is an explicit incorporation of uncertainty in results.

The most notable difference between the accidents with respect to the source term is the failure mode of the emergency core cooling system (ECCS). In two of the cases, both of the reactor recirculation valves (RRVs) open while all three reactor vent valves (RVVs) fail to open. In this type of accident, there is return of coolant from the containment vessel (CNV) to the reactor pressure vessel (RPV) shortly after core damage and the source term is very small. In the other three cases, either the RRVs fail to open while all three RVVs open, or all five ECCS valves fail to open. This ECCS failure results in a permanent loss of coolant from the RPV and more significant core damage. Therefore, there is a bi-modal distribution in source terms from the five severe accidents, with two on the low end and three on the high end of release fraction. Using the median source term ensures the core damage source term (CDST) is taken from the three larger releases and is notably conservative when compared to a mean source term.

The source term to containment from the three accidents with greater release was largely consistent. Had one of the results been significantly higher, this would have prompted investigation into the cause of the divergence and potentially resulted in the use of the largest source term to account for uncertainty. The similar results among the three cases provides confidence that the median result was a reasonable representation of the spectrum of severe accidents.

In the report referenced by the staff (RES/FSCB 2019-01) results from three severe accidents were compared between the staff's simulations and the results presented in NuScale's FSAR Section 19.2. While the RAI basis cites one comparison in which the NRC's results were greater, {{

}}[2(a),(c)] The result referenced in the RAI basis was the outlier.

{{

}}[2(a),(c)] Late in an accident, when the RPV and CNV have reached equilibrium pressure and hydrogen production has halted, there is no motive force to carry radionuclides into the CNV and radionuclide deposition in the RPV is the most probable outcome. Increased release from fuel does not necessarily correspond to an increase in release to the containment.

Another significant justification for the relatively lower releases in the CDST is the success of the DHRS. All five of the severe accidents that comprise the CDST had successful operation of both trains of DHRS. As shown in NuScale's PRA, having successful DHRS is the most likely outcome, especially since the two trains are fully redundant and either is capable of removing one hundred percent of decay heat. Success of DHRS slows the accident progression and extends the time until the core is uncovered. With less decay heat remaining when core damage occurs, it is expected that the severity of core damage is less and the releases are lower. This is in comparison to the six simulations in RES/FSCB 2019-01 which all had failure of DHRS.

{{

}}2(a),(c)

There is no requirement for the source term to be demonstrably conservative in every step. There are significant conservatisms in the dose calculations which use this source term as input. For example, the entire release is condensed into a one hour duration, although the actual release may be over 48 hours. As specified by R.G. 1.183, "*The AST…must represent a spectrum of credible severe accident events*". As evidenced by this response, the use of a median result from five accidents with a range of failures meets the guidance of R.G. 1.183 and has appropriate consideration of uncertainties. The best estimate results used as input to dose calculations are demonstrably representative. Therefore it would be inappropriate to speculate as to whether any dose criteria would be exceeded if the source term was arbitrarily increased.

Impact on Topical Report:

There are no impacts to the Topical Report TR-0915-17565, Accident Source Term Methodology, as a result of this response.

September 05, 2019

Docket: PROJ0769

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

SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 9690 (eRAI No. 9690) on the NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3

REFERENCES:
1. U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 9690 (eRAI No. 9690)," dated June 27, 2019
2. NuScale Topical Report, "Accident Source Term Methodology," TR-0915-17565, Revision 3, dated April 2019
3. NuScale Power, LLC Response to NRC "Request for Additional Information No. 9690 (eRAI No.9690)," dated July 31, 2019

The purpose of this letter is to provide the NuScale Power, LLC (NuScale) response to the referenced NRC Request for Additional Information (RAI).

The Enclosure to this letter contains NuScale's response to the following RAI Question from NRC eRAI No. 9690:

- 01.05-40

The responses to RAI Questions 01.05-39, 01.05-41 and 01.05-42 were previously provided in Reference 3. This completes all responses to eRAI 9690.

This letter and the enclosed response make no new regulatory commitments and no revisions to any existing regulatory commitments.

If you have any questions on this response, please contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com.

Sincerely,

Zackary W. Rad
Director, Regulatory Affairs
NuScale Power, LLC

Distribution: Gregory Cranston, NRC, OWFN-8H12
Samuel Lee, NRC, OWFN-8H12
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9690

NuScale Power, LLC
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Enclosure 1:

NuScale Response to NRC Request for Additional Information eRAI No. 9690

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NUSCALE
Power for all humankind

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9690

Date of RAI Issue: 06/27/2019

NRC Question No.: 01.05-40

Regulatory Basis:

10 CFR 52.47(a)(2)(iv) requires that an application for a design certification include a final safety analysis report that provides a description and safety assessment of the facility. The safety assessment analyses are completed, in part, to show compliance with the radiological consequence evaluation factors in 52.47(a)(2)(iv)(A) and 52.47(a)(2)(iv)(B) for offsite doses; and 10 CFR Part 50, Appendix A, GDC 19, 10 CFR 50.34(f)(2)(vii) and 10 CFR 50.34(f)(2)(xxviii) for control room radiological habitability. The radiological consequences of design basis accidents are evaluated against these regulatory requirements and the dose acceptance criteria given in NuScale design specific review standard (DSRS) Section 15.0.3. Regulatory Guide 1.183 provides dose assessment guidance.

Background:

On January 31, 2019, NuScale submitted a request for exemption from the 10 CFR 50.34(f)(2)(viii) requirements related to post-accident sampling. In the technical basis for the exemption request, the applicant describes that the capability to continuously monitor hydrogen and oxygen concentration in the containment atmosphere to meet the requirements of 10 CFR 50.44 is accomplished using the process sampling system (PSS) in-line monitors during accident conditions, including beyond-design-basis events with core damage. The path that the highly radioactive post-accident containment atmosphere would take to achieve continuous monitoring of hydrogen and oxygen concentration is outside of the containment, and the PSS is not related to safety. NuScale topical report TR- 0915-17565, Revision 3, "Accident Source Term Methodology," was submitted on April 21, 2019. This topical report describes the accident source term and radiological consequence analysis methodology for the core damage event (CDE), which is used to show compliance with the regulatory requirements described above.

The description of the CDE radiological consequence analysis in Section 4.2.5 of the topical report does not include discussion of the potential releases from the post-accident combustible gas monitoring pathway outside containment.

RG 1.183, Appendix A provides guidance on modeling of potential pathways to the environment for core damage accidents in the radiological consequence analyses which show compliance with the regulations stated above. Guidance on the modeling of ESF system leakage and containment purging in RG 1.183, although not directly describing the NuScale design post-accident combustible gas monitoring capability, provides indication that potential release pathways to the environment for the accident should be included in the analysis.

Issue:

Additional information is needed to describe the modeling of potential releases to the environment through the systems used in post-accident monitoring of hydrogen and oxygen concentration in the containment atmosphere to demonstrate compliance with the regulatory requirements described above.

Request:

Please describe the methods, models, and assumptions used for calculating the contribution to the dose from a potential release to the environment through leakage from the systems used in post-accident monitoring of the hydrogen and oxygen concentration in the containment atmosphere for the CDE. Additionally, please update the topical report to provide this description and make any necessary conforming changes to the FSAR.

NuScale Response:

The hydrogen and oxygen monitoring capability is provided in the NuScale design as part of the process sampling system (PSS), via the containment evacuation system (CES) and the core flood and drain system (CFDS). Unlike the engineered safety feature (ESF) systems, for which leakage is addressed by Regulatory Guide 1.183, the hydrogen and oxygen monitoring capability is not relied upon for mitigating any design basis event and is not credited in any design basis event analysis. Combustible gas monitoring is provided pursuant to 10 CFR 50.44(c)(4), which requires combustible gas monitoring capability only for beyond-design basis events (reference 68 FR 54126).

Regulatory Guide 1.183, Appendix A, Section 7 states, in part, “If the installed containment purging capabilities are maintained for purposes of severe accident management and are not credited in any design basis analysis, radiological consequences need not be evaluated.” As with severe accident containment purge capability, the NuScale hydrogen and oxygen monitoring capability is only provided for the purpose of severe accident management, and is not credited in any design basis analysis. Therefore, as with the radiological consequences of a containment purge for combustible gas control (a large release), evaluation of the radiological consequences of a potential combustible gas monitoring leak is unnecessary (a small release by comparison).

Notwithstanding the Regulatory Guide 1.183 position on severe accident management doses, NuScale recognizes that the Core Damage Event (CDE) analyzed in FSAR section 15.10 is a beyond-design-basis accident for which NuScale has elected to analyze dose consequences in comparison to offsite and control room dose limits. NRC Staff have stated this analysis implies that all potential dose contributors, including those from the combustible gas monitoring function required by 10 CFR 50.44 for severe accidents, must be accounted for. However, NuScale notes that the CDE is analogous to the “design basis loss of coolant accident” (DBLOCA) radiological consequence analysis prescribed by SRP section 15.6.5, which also assumes a significant core damage event, and in turn would seem to necessitate the use of combustible gas monitoring pursuant to 10 CFR 50.44. Also, all plants are required to have a capability for post-accident sampling following an accident. However, SRP 15.6.5 and Regulatory Guides 1.183 and 1.195 address neither combustible gas monitoring leakage nor post-accident sampling releases and leakage.

Three Mile Island Action Plan Item III.D.1.1.1, applicable to NuScale by way of 10 CFR 50.34(f)(2)(xxvi), addresses potential leakage from systems such as the combustible gas monitoring equipment that may contain accident source term following a core damage accident. Licensees are required to have a leakage control program including actions to minimize leakage from such systems, where the “goal is to minimize potential exposures to workers and public, and to provide reasonable assurance that excessive leakage will not prevent the use of systems needed in an emergency.” NUREG-0737 indicates that implementation of the leakage control program provides reasonable assurance that doses from such systems is acceptable, without specifically calculating the dose contribution from hypothetical leakage. For example, Item II.B.2 provides that, in calculating the dose to operators in conducting post-accident sampling, “radiation from leakage of systems located outside of containment need not be considered” because “leakage measurement and reduction is treated under Item III.D.I.I.” Similarly, Item III.D.3.4 prescribes that the control room dose analysis should be for containment leakage and ESF system leakage outside containment. In sum, NUREG-0737 identifies potential leakage outside containment as an issue to address by reducing leakage under Item III.D.1.1, but omits this known, potential leakage source as a source term to be evaluated for doses to operators under Items II.B.2 and III.D.3.4.

Therefore, NuScale’s position is that implementation of a leakage control program pursuant to 10 CFR 50.34(f)(2)(xxvi) and NUREG-0737 item III.D.1.1 demonstrates compliance with the regulatory requirements Staff identified, without modeling the dose consequences of hypothetical leakage from combustible gas monitoring. NUREG-0737 item III.D.1.1 requires that a licensee “reduce leakage to as-low-as-practical levels.”

Precedent supports NuScale's position. For example, the ESBWR design includes a Containment Monitoring System (CMS) that is partially routed outside of containment (ESBWR DCD, Chapter 7, Figure 7.5-1, Rev. 10; ML14100A523) and is used to monitor the hydrogen and oxygen gas concentrations in the drywell and wetwell during post-accident conditions, however there is no leakage assumed from the CMS in the accident dose evaluations (ESBWR DCD, Chapter 15, Rev. 10; ML14100A547), and the leakage control program specifies that leakage from that and other systems containing accident source term outside containment be “as low as practicable” (ESBWR DCD, Chapter 16, Generic Technical Specification 5.5.2). The APR1400 also includes a hydrogen monitor outside containment as part of its CMS (APR1400 DCD, Section 6.2.5.2.2), which is likewise included in scope of the leakage control program without quantifying an acceptance criterion (APR1400 DCD, Chapter 16, Generic Technical Specification 5.5.2). The ABWR includes 13 systems outside containment that may contain source term following an accident, including post-accident sampling, process sampling, containment atmospheric monitoring, fission product monitoring (ABWR DCD, Section 1A.2.34, Rev. 4). Such systems are subject to the leakage control program but leakage from them does not appear to be calculated in accident dose evaluations. The ABWR SER concludes that a requirement for COL applicant’s procedures to “reduce detected leakages to lowest practical levels” satisfied TMI item III.D.1.1 (NUREG-1503, Section 20.5.38).

In the NuScale design, the combustible gas monitoring loop is a gaseous process stream, and the system is used during normal operations. Substantial system leakage during normal operation would be evident because it would inhibit the ability to maintain a vacuum inside containment. Also, the combustible gas monitoring loop is included in the leakage control program pursuant to COL Item 9.3-1 in order to "minimize potential exposures to workers and the public, and provide reasonable assurance that excessive leakage will not prevent the system's use in an emergency" (10 CFR 50.34(f)(2)(xxvi)). Consistent with precedent and NUREG-0737 item III.D.1.1, COL Item 9.3-1 will be updated to explicitly identify "as low as practical" as the acceptance criterion for the leakage control program, and to identify as within program scope the systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere. A COL licensee's procedures will implement the leakage reduction program consistent with this requirement.

In a recent public meeting, NRC Staff indicated that the ESBWR precedent is not applicable to the NuScale design because the ESBWR has CMS isolation valves inside containment and can isolate the CMS from the control room. NuScale does not believe these differences are relevant. Leakage up to and including that from the NuScale CES and CFDS containment isolation valves is included in the allowable and measured containment leakage. Isolation of the ESBWR CMS would appear to render combustible gas monitoring non-functional in contravention of 10 CFR 50.44, as NRC Staff earlier stated would be the case for NuScale, and the location of isolation is not relevant to offsite and control room dose (local operator dose is addressed in RAI 9682, Q12.03-66).

Therefore, NuScale has not performed a radiological consequence analysis of a potential leak from the beyond design basis post-accident hydrogen and oxygen monitoring process.

Impact on DCA:

FSAR Section 9.3.2 and Table 1.8-2 have been revised as described in the response above and as shown in the markup provided in this response.

RAI 01-61, RAI 01.05-40, RAI 02.04.13-1, RAI 03.04.01-4, RAI 03.04.02-1, RAI 03.04.02-2, RAI 03.04.02-3, RAI 03.05.01.03-1, RAI 03.05.01.04-1, RAI 03.05.02-2, RAI 03.05.03-4, RAI 03.06.02-6, RAI 03.06.02-15, RAI 03.06.03-11, RAI 03.07.01-2, RAI 03.07.01-3, RAI 03.07.02-4S3, RAI 03.07.02-6S1, RAI 03.07.02-6S2, RAI 03.07.02-8, RAI 03.07.02-12, RAI 03.07.02-15S5, RAI 03.07.02-16S1, RAI 03.07.02-23S1, RAI 03.07.02-26, RAI 03.08.04-1S1, RAI 03.08.04-3S2, RAI 03.08.04-23S1, RAI 03.08.04-23S2, RAI 03.08.04-23S3, RAI 03.08.05-14S1, RAI 03.09.02-15, RAI 03.09.02-48, RAI 03.09.02-67, RAI 03.09.02-69, RAI 03.09.03-12, RAI 03.09.06-5, RAI 03.09.06-6, RAI 03.09.06-16, RAI 03.09.06-16S1, RAI 03.09.06-27, RAI 03.11-8, RAI 03.11-14, RAI 03.11-14S1, RAI 03.11-18, RAI 03.11-19S2, RAI 03.13-3, RAI 04.02-1S2, RAI 05.02.03-19, RAI 05.02.05-8, RAI 05.04.02.01-13, RAI 05.04.02.01-14, RAI 05.04.02.01-19, RAI 06.02.01.01.A-18, RAI 06.02.01.01.A-19, RAI 06.02.06-22, RAI 06.02.06-23, RAI 06.04-1, RAI 09.01.01-20, RAI 09.01.01-20S1, RAI 09.01.02-4, RAI 09.01.05-3, RAI 09.01.05-6, RAI 09.03.02-2S1, RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8, RAI 10.02-1, RAI 10.02-2, RAI 10.02-3, RAI 10.02.03-1, RAI 10.02.03-2, RAI 10.03.06-1, RAI 10.03.06-5, RAI 10.04.06-1, RAI 10.04.06-2, RAI 10.04.06-3, RAI 10.04.10-2, RAI 11.01-2, RAI 11.01-2S2, RAI 12.03-55S1, RAI 12.03-63, RAI 13.01.01-1, RAI 13.01.01-1S1, RAI 13.02.02-1, RAI 13.03-4, RAI 13.05.02.01-2, RAI 13.05.02.01-2S1, RAI 13.05.02.01-3, RAI 13.05.02.01-3S1, RAI 13.05.02.01-4, RAI 13.05.02.01-4S1, RAI 14.02-7, RAI 16-65, RAI 19-31, RAI 19-31S1, RAI 19-38, RAI 20.01-13

Table 1.8-2: Combined License Information Items

Item No.	Description of COL Information Item	Section
COL Item 1.1-1:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific plant location.	1.1
COL Item 1.1-2:	A COL applicant that references the NuScale Power Plant design certification will provide the schedules for completion of construction and commercial operation of each power module.	1.1
COL Item 1.4-1:	A COL applicant that references the NuScale Power Plant design certification will identify the prime agents or contractors for the construction and operation of the nuclear power plant.	1.4
COL Item 1.7-1:	A COL applicant that references the NuScale Power Plant design certification will provide site-specific diagrams and legends, as applicable.	1.7
COL Item 1.7-2:	A COL applicant that references the NuScale Power Plant design certification will list additional site-specific piping and instrumentation diagrams and legends as applicable.	1.7
COL Item 1.8-1:	A COL applicant that references the NuScale Power Plant design certification will provide a list of departures from the certified design.	1.8
COL Item 1.9-1:	A COL applicant that references the NuScale Power Plant design certification will review and address the conformance with regulatory criteria in effect six months before the docket date of the COL application for the site-specific portions and operational aspects of the facility design.	1.9
COL Item 1.10-1:	A COL applicant that references the NuScale Power Plant design certification will evaluate the potential hazards resulting from construction activities of the new NuScale facility to the safety-related and risk significant structures, systems, and components of existing operating unit(s) and newly constructed operating unit(s) at the co-located site per 10 CFR 52.79(a)(31). The evaluation will include identification of management and administrative controls necessary to eliminate or mitigate the consequences of potential hazards and demonstration that the limiting conditions for operation of an operating unit would not be exceeded. This COL item is not applicable for construction activities (build-out of the facility) at an individual NuScale Power Plant with operating NuScale Power Modules.	1.10
COL Item 2.0-1:	A COL applicant that references the NuScale Power Plant design certification will demonstrate that site-specific characteristics are bounded by the ~~design~~site parameters specified in Table 2.0-1. If site-specific values are not bounded by the values in Table 2.0-1, the COL applicant will demonstrate the acceptability of the site-specific values in the appropriate sections of its combined license application.	2.0
COL Item 2.1-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site geographic and demographic characteristics.	2.1
COL Item 2.2-1:	A COL applicant that references the NuScale Power Plant design certification will describe nearby industrial, transportation, and military facilities. The COL applicant will demonstrate that the design is acceptable for each ~~potential accident~~of these potential hazards, or provide site-specific design alternatives.	2.2
COL Item 2.3-1:	A COL applicant that references the NuScale Power Plant design certification will describe the site-specific meteorological characteristics for Section 2.3.1 through Section 2.3.5, as applicable.	2.3

Table 1.8-2: Combined License Information Items (Continued)

Item No.	Description of COL Information Item	Section
COL Item 9.1-9:	A COL applicant that references the NuScale Power Plant design certification will provide a neutron absorber material qualification report which demonstrates that the neutron absorber material can meet the neutron attenuation and environmental compatibility design functions described in Technical Report TR-0816-49833. The COL applicant will establish procedures to evaluate the neutron attenuation uncertainty associated with the material lot variability and will establish procedures to inspect the as-manufactured material for contamination and manufacturing defects.	9.1
COL Item 9.2-1:	A COL applicant that references the NuScale Power Plant design certification will select the appropriate chemicals for the reactor component cooling water system based on site-specific water quality and materials requirements.	9.2
COL Item 9.2-2:	A COL applicant that references the NuScale Power Plant design certification will describe the source and pre-treatment methods of potable water for the site, including the use of associated pumps and storage tanks.	9.2
COL Item 9.2-3:	A COL applicant that references the NuScale Power Plant design certification will describe the method for sanitary waste storage and disposal, including associated treatment facilities.	9.2
COL Item 9.2-4:	A COL applicant that references the NuScale Power Plant design certification will provide details on the prevention of long-term corrosion and organic fouling in the site cooling water system.	9.2
COL Item 9.2-5:	A COL applicant that references the NuScale Power Plant design certification will identify the site-specific water source and provide a water treatment system that is capable of producing water that meets the plant water chemistry requirements.	9.2
COL Item 9.3-1:	A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). The leakage control program will include~~, including~~ an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.	9.3
COL Item 9.3-2:	Not used.	9.3
COL Item 9.4-1:	A COL applicant that references the NuScale Power Plant design certification will specify a periodic testing and inspection program for the normal control room heating ventilation and air conditioning system.	9.4
COL Item 9.4-2:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Reactor Building heating ventilation and air conditioning system in accordance with Regulatory Guide 1.140.	9.4
COL Item 9.4-3:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Radioactive Waste Building heating ventilation and air conditioning system.	9.4
COL Item 9.4-4:	A COL applicant that references the NuScale Power Plant design certification will specify periodic testing and inspection requirements for the Turbine Building heating ventilation and air conditioning system.	9.4
COL Item 9.5-1:	A COL applicant that references the NuScale Power Plant design certification will provide a description of the offsite communication system, how that system interfaces with the onsite communications system, as well as how continuous communications capability is maintained to ensure effective command and control with onsite and offsite resources during both normal and emergency situations.	9.5
COL Item 9.5-2:	A COL applicant that references the NuScale Power Plant design certification will determine the location for the security power equipment within a vital area in accordance with 10 CFR 73.55(e)(9)(vi)(B).	9.5
COL Item 10.2-1:	Not used.	10.2
COL Item 10.2-2:	Not used.	10.2
COL Item 10.2-3:	Not used.	10.2

Consistent with 10 CFR 50.34(f)(2)(xvii)(c) and 10 CFR 50.44(c)(4) the PSS design provides equipment capable of continuous monitoring of hydrogen and oxygen concentration in the containment atmosphere. The equipment used for monitoring hydrogen is reliable and capable of continuously measuring the concentration of hydrogen in the containment atmosphere following a significant beyond design basis accident for accident management and provides indication in the MCR.

RAI 01.05-40

Consistent with 10 CFR 50.34(f)(2)(xxvi), the PSS design contains provisions for leakage detection, and to control leakage to levels as low as practical~~control and detection~~, to minimize exposures to workers and the public and to maintain control and use of the system during an accident (Item III.D.1.1 in NUREG-0737).

RAI 01.05-40

COL Item 9.3-1: A COL applicant that references the NuScale Power Plant design certification will submit a leakage control program for systems outside containment that contain (or might contain) accident source term radioactive materials following an accident (including systems and components used in post-accident hydrogen and oxygen monitoring of the containment atmosphere). The leakage control program will include~~, including~~ an initial test program, a schedule for re-testing these systems, and the actions to be taken for minimizing leakage from such systems to as low as practical.

Consistent with 10 CFR 20.1101(b), the PSS design supports keeping radiation exposures as low as reasonably achievable (ALARA). Consistent with 10 CFR 20.1406, the PSS design supports minimization of contamination of the facility and the environment, minimizing generation of radioactive waste, and facilitating eventual plant decommissioning.

9.3.2.2 System Description

9.3.2.2.1 General Description

The PSS is designed to collect representative liquid and gaseous samples from various plant systems using the following sampling system features:

- the primary sampling system
- the containment sampling system (CSS)
- the secondary sampling system (SSS)
- local grab sample provisions

The PSS is operable during normal operations, including at power, shutdown, and startup. The system has the ability to obtain samples at the normal system operating temperatures and pressures from various locations. These samples can be in the form of continuously analyzed samples or grab samples. The PSS obtains samples that are representative of the process or vessel under evaluation. For sampling of process streams, sample points are located in a turbulent flow zone which minimizes particulate dropout and re-entrainment in sample piping. For addition, a break in a sample line would result in activity release that might actuate the fixed area radiation monitors located in the containment sampling system equipment area and the primary sampling system equipment area, as described in Table 12.3-10. The three PSS sample points to the CVCS are each provided with two fail-closed isolation valves. These isolation valves are downstream of the environmentally qualified CIVs associated with the CVCS discharge line and are also downstream of the CVCS module isolation valves as shown on Figure 9.3.4-1. The PSS line sizes range from 3/4" to 3/8" which further restricts the break flow of a sample line outside containment.

The PSS design satisfies GDC 63 by allowing the detection of conditions that may result in excessive radiation levels in the fuel storage and radioactive waste systems. The PSS includes sampling capability of the spent fuel pool and reactor pool water via local sample points in the pool cooling and cleanup system. The PSS also includes sampling capability via local sample points for the radioactive waste management systems. This enables analyses to be performed to detect conditions in the fuel storage and radioactive waste systems that could result in excessive radiation levels and excessive personnel exposure.

RAI 09.03.02-3S1

The PSS design satisfies GDC 64 as it provides the capability to sample and analyze for radioactivity that may be released during normal operations, anticipated operational occurrences, and postulated accidents.

RAI 09.03.02-3, RAI 09.03.02-3S1, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The PSS design satisfies 10 CFR 50.34(f)(2)(xvii)(c) by providing capability to monitor hydrogen and oxygen concentration in containment atmosphere during operation and during beyond design-basis conditions. The monitor is a nonsafety-related instrument that sends output signal to the MCS to provide readout in the main control room.

RAI 01.05-40, RAI 09.03.02-3, RAI 09.03.02-4, RAI 09.03.02-5, RAI 09.03.02-6, RAI 09.03.02-8

The PSS design satisfies 10 CFR 50.34(f)(2)(xxvi) (Item III.D.1.1 in NUREG-0737), as it relates to including provisions for leakage control and detection to levels as low as practical to prevent unnecessarily high exposures to workers and the public and to maintain control and use of the system post-accident. The PSS design includes provisions for leakage control and detection. Flow and pressure instrumentation on the sample lines can provide indication of potential leaks. Radiation monitoring capabilities are provided for detecting excessive radiation level resulting from system leakage. The sample line can be isolated upon detection of high radiation by the CVCS or CES process radiation monitor located upstream of the sample line as shown in Figure 9.3.4-1 and Figure 9.3.6-1 respectively. Excessive radiation level detected by the fixed area radiation monitor located in the primary sampling system or the containment sampling system equipment areas described in Table 12.3-10 can also provide indication of system leakage that warrants system isolation for leakage control.

The PSS design satisfies the requirements of 10 CFR 50.44(c)(4), as the equipment design attributes conform to RG 1.7 regulatory position C.2. It provides the ability to monitor containment hydrogen and oxygen using an in-line monitor for both normal and accident conditions. In addition grab sampling provisions are provided on the CES

Section D

April 21, 2019 Docket No. PROJ0769

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

SUBJECT: NuScale Power, LLC Submittal of "Accident Source Term Methodology," TR-0915-17565, Revision 3

REFERENCES: Letter from NuScale Power, LLC to Nuclear Regulatory Commission, "NuScale Power, LLC Submittal of Topical Report "Accident Source Term Methodology, TR-0915-17565, Revision 2," dated September 11, 2017 (ML17254B0468)

NuScale Power, LLC (NuScale) hereby submits Revision 3 of the "Accident Source Term Methodology" (TR-0915-17565) topical report. The purpose of this submittal is to request that the NRC review and approve NuScale's accident source term methodology.

Enclosure 1 contains the proprietary version of the report entitled "Accident Source Term Methodology." NuScale requests that the proprietary version be withheld from public disclosure in accordance with the requirements of 10 CFR § 2.390. The enclosed affidavit (Enclosure 3) supports this request. Enclosure 2 contains the nonproprietary version of the report entitled "Accident Source Term Methodology."

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

If you have any questions, please feel free to contact Carrie Fosaaen at 541-452-7126 or at cfosaaen@nuscalepower.com if you have any questions.

Sincerely,

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

Distribution: Samuel Lee, NRC, OWFN-8H12
Gregory Cranston, NRC, OWFN-8H12

Enclosure 1: "Accident Source Term Methodology," TR-0915-17565-P, Revision 3, proprietary version
Enclosure 2: "Accident Source Term Methodology," TR-0915-17565-NP, Revision 3, nonproprietary version
Enclosure 3: Affidavit of Thomas A. Bergman, AF-0419-65236

Enclosure 2:

"Accident Source Term Methodology," TR-0915-17565, Revision 3, nonproprietary version

Note: NuScale submitted TR-0915-17565, "Accident Source Term Methodology," Revision 3 on April 21, 2019 (ML#19112A172). At the direction of the NRC, NuScale submitted updates to Revision 3 in a letter dated July 31, 2019 (ML#19212A802). The updates, together with Revision 3, became known as Revision 4 to the Accident Source Term Methodology topical report. Section B of this submittal is the full Revision 4 report with the –A designation added. Therefore, Revision 3 is not included in this enclosure.

NuScale Power, LLC
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Enclosure 3:

Affidavit of Zackary W. Rad, AF-0220-68979

NuScale Power, LLC

AFFIDAVIT of Zackary W. Rad

I, Zackary W. Rad, state as follows:

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

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

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

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

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

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

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

(3) Public disclosure of the information sought to be withheld is likely to cause substantial harm to NuScale's competitive position and foreclose or reduce the availability of profit-making opportunities. The accompanying report reveals distinguishing aspects about the methodology by which NuScale develops its accident source term.

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

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

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

(4) The information sought to be withheld is in Enclosure 1 to the "NuScale Power, LLC Submittal of the Approved Version of the NuScale Topical Report 'Accident Source Term Methodology.' TR-0915-17565, Revision 3." The enclosure contains the designation "Proprietary" at the top of each page containing proprietary information. The information considered by NuScale to be proprietary is identified within double braces, "{{ }}" in the document.

(5) The basis for proposing that the information be withheld is that NuScale treats the information as a trade secret, privileged, or as confidential commercial or financial information. NuScale relies upon

the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC § 552(b)(4), as well as exemptions applicable to the NRC under 10 CFR §§ 2.390(a)(4) and 9.17(a)(4).

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

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

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

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

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

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

I declare under penalty of perjury that the foregoing is true and correct. Executed on February 26, 2020.

Zackary W. Rad