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 23, 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 (99.5^{th}-percentile concentrations) for each downwind sector and a relative concentration that is exceeded no more than 5 percent of the time (95^{th}-percentile concentration) for all sectors combined in one run, whereas ARCON96 calculates a 95^{th}-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.

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4.2.1.6 Description of NuScale's ARCON96 Methodology

Sections 4.1.6 and 6.1.1 of the topical report state 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.
3. U.S. Nuclear Regulatory Commission, RG 1.183, “Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors,” July 2000.
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