RAIO-1217-57597

December 11, 2017

Docket No. 52-048

U.S. Nuclear Regulatory Commission
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11555 Rockville Pike
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SUBJECT: NuScale Power, LLC Response to NRC Request for Additional Information No. 270 (eRAI No. 9185) on the NuScale Design Certification Application

REFERENCE: U.S. Nuclear Regulatory Commission, "Request for Additional Information No. 270 (eRAI No. 9185)," dated October 23, 2017

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. 9185:

- 02.03.04-1

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 Marty Bryan at 541-452-7172 or at mbryan@nuscalepower.com.

Sincerely,

Zackary W. Rad
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Enclosure 1: NuScale Response to NRC Request for Additional Information eRAI No. 9185

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NuScale Response to NRC Request for Additional Information eRAI No. 9185

Response to Request for Additional Information
Docket No. 52-048

eRAI No.: 9185
Date of RAI Issue: 10/23/2017

NRC Question No.: 02.03.04-1

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). 10 CFR Part 50, Appendix A, General Design Criterion 19 (GDC 19), "Control Room," requires an evaluation of personnel exposures inside the control room during radiological accident conditions.

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. RG 1.194, “Atmospheric Relative Concentrations for Control room Radiological Habitability Assessments at Nuclear Power Plants,” presents criteria for characterizing atmospheric dispersion conditions for evaluating the consequences of radiological releases to the control room. RG 1.194 states the use of the ARCON96 computer code (NUREG/CR- 6331, Rev. 1) is an acceptable approach for implementing some of the criteria for evaluating the consequences of airborne radiological releases to the control room.

Information Request Background

NuScale Topical Report TR-0915-17565-P, Revision 2, “Accident Source Term Methodology,” (ML17254B067) describes the methodology used for establishing source terms and calculating the atmospheric dispersion factors (χ/Q values) used to determine accident radiological consequences at the technical support center (TSC), main control room (MCR) and offsite locations for the NuScale Power Plant certified design. This topical report is also intended to provide guidance to COL applicants in determining their site-specific χ/Q values for comparison with the NuScale site parameter χ/Q values.

Offsite χ/ Q Values

Background

SRP 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accident Releases,” states that the site parameters postulated by a DC applicant should be representative for a reasonable number of sites that have been or may be considered for a COL applicant. NuScale design certification application (DCA) FSAR Tier 1, Table 5.0-1, and Tier 2, Table 2.0-1 provide accident release χ/Q site parameter values for the security owner controlled area fence. NuScale DCA FSAR, Tier 2, Section 2.3.4 states this fence may be used as the EAB and LPZ outer boundary.

Section 5 of Topical Report TR-0915-17565-P provides example calculation analyses and results to demonstrate the application of the accident release χ/Q methodology described in the Topical Report. Three years of meteorological data from Sacramento, California were used in the example χ/Q calculations provided in the report. The Sacramento meteorological data were chosen because a study of data from 241 National Weather Service stations across the U.S. showed this site represented an 80—90th percentile U.S. site. The resulting offsite χ/Q values, shown in Table 5-4 of TR-0915-17565-P, are the same χ/Q values used as the accident release security owner controlled area fence site parameter values presented in the DCA FSAR.

In response to NuScale Topical Report (TR-0915-17565-P) RAI 8881, Question 02.03.04-2 (ML17236A528), the applicant revised the methodology for calculating offsite χ/Q values in Revision 2 to the Topical Report, to more fully conform to the more conservative RG 1.145 guidance regarding the calculation of maximum sector and overall site χ/Q values. However, the resulting offsite χ/Q values shown in Table 5-4 of Topical Report TR-0915- 17565-P, Revision 2 were not recalculated to represent the more conservative methodology incorporated into Topical Report TR-0915-17565-P, Revision 2.

Information Request

a. Please justify why the NuScale accident release security owner controlled area fence χ/Q site parameter values presented in the NuScale DCA FSAR Tier 1, Table 5.0-1, and Tier 2, Table 2.0-1 were not revised to reflect the more conservative atmospheric dispersion methodology incorporated into Revision 2 of Topical Report TR-0915-17565-P.
b. If the NuScale accident release security owner controlled area fence χ/Q site parameter values are revised and reflect more limiting χ/Q site parameter values, please describe the impacts on and update the corresponding radiological accidents evaluated in the NuScale DCA FSAR.

Onsite χ/Q Value

Background

SRP 2.3.4, “Short-Term Atmospheric Dispersion Estimates for Accident Releases,” states that a DC application should contain figures and tables showing the design features that would be used by the COL applicant to generate MCR χ/Q values (e.g., intake heights, release heights, building cross-sectional areas, and distances to receptors).

NuScale DCA FSAR Tier 2, Table 2.0-1 provides some release point characteristic assumptions used for the NuScale TSC and MCR site parameter χ/Q calculations. NuScale DCA FSAR Tier 2, Figures 2.3-2 and 2.3-3 also show the path and distances from the reactor building release points to the MCR door and HVAC inlet, respectively.

Information Request

a. Please revise the DCA FSAR to include all the NuScale nuclear plant configuration data required by COL applicants to perform atmospheric dispersion modeling (e.g., release heights, intake heights, building area).
b. Please revise the directions shown in NuScale DCA FSAR Tier 2, Figures 2.3-2 and 2.3- 3 from “source to receptor” directions to ”receptor to source” directions.

c. Please provide the basis that supports NuScale’s position that the two source locations shown in NuScale DCA FSAR Tier 2, Figures 2.3-2 and 2.3-3 (e.g., the reactor building SE and NE single personnel doors) are the limiting source locations for all radiological accidents evaluated in the NuScale DCA FSAR.

NuScale Response:

First Information Request

a. The X/Q site parameter values presented in FSAR Tier 1, Table 5.0-1 and Tier 2, Table 2.0-1 were revised as described in LO-1117-57037 (ML17317B546).

b. Changes to the X/Q site parameter values affect the radiological accident calculated dose results in a linear manner. LO-1117-57037 (ML17317B546) shows the revised radiological accident calculated dose results. Many other parameters besides X/Q affect radiological accident calculated dose results and some of these parameters have changed. Therefore, the results have not changed in a linear manner.

Second Information Request

a. Pertinent parameters were moved from FSAR Tier 2, Table 2.0-1 to Tables 11.3-12 and 15.0-20 and "intake height" was added as a parameter, as shown in the attached markup.

b. FSAR Tier 2, Figures 2.3-2 and 2.3-3 are revised as requested, as shown in the attached markup. Note that the degrees are no longer shown. This is because the degree input to NARCON is based on the angle between the source and the receptor in relation to the true north which the weather data is based on. Since the layout of the buildings in the FSAR are in relation to an arbitrary "site north" which may or may not align with the as constructed plant's orientation to true north, this NARCON input can not be generically stated and the COL applicant will have to establish the correct input to NARCON if they have to redo the control room dose analysis.

c. The two source locations shown in Figures 2.3-2 and 2.3-3 are the limiting source locations because they are the closest source locations to the main control room personnel doors and main control room HVAC intake. FSAR Tier 2, Section 2.3.4 has been updated as shown in the attached markup to clarify this.

Impact on DCA:

FSAR Sections 2.3, 11.3, and 15.0 have been revised as described in the response above and as shown in the markup provided in this response.

2.3.4 Short-Term Atmospheric Dispersion Estimates for Accident Releases

Accidental Radioactive Releases

Topical Report TR-0915-17565, Revision 0, (Reference 2.3-3) describes the methodology used for establishing source terms and calculating the atmospheric dispersion factors used to determine accident radiological consequences at the technical support center (TSC), main control room (MCR) and offsite locations for the NuScale Power Plant certified design.

Atmospheric dispersion factors (χ/Q values) are determined at the site owner controlled area boundary. This fence is 400 feet from the closest release point and may be used as both the exclusion area boundary (EAB) and as the low population zone (LPZ) outer boundary. These χ/Q values as well as the χ/Q values for the MCR were determined for various sites in the United States using a meteorological database that included multiple years of data across all regions of the United States. This approach determined that the meteorological dataset for Sacramento, California, between 1984-1986, is representative of the bounding 80th to 90th percentile of potential NuScale Power Plant construction sites in the United States. This meteorological data set was used to calculate the χ/Q values for the certified design.

The χ/Q values at the site owner controlled area fence are listed in Table 2.0-1. These χ/Q values are based on the source location and path shown in Figure 2.3-1.

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The χ/Q values used for evaluation of doses in the MCR and TSC are determined at the Control Building doors and HVAC inlet and are listed in Table 2.0-1. Figure 2.3-2 and Figure 2.3-3 show the path and distances from the Reactor Building release point to MCR door and HVAC inlet. The two source locations shown in Figure 2.3-2 and Figure 2.3-3 are the limiting source locations because they are the closest source locations to the main control room personnel doors and main control room HVAC intake. Assumptions for release point characteristics used for the χ/Q calculations are ~~also~~ listed in ~~Table 2.0-1.~~Table 15.0-20.

The χ/Q values for the TSC are the same as the MCR because the TSC is located directly above the MCR and shares the same HVAC inlet and outside doors.

The COL applicant will determine site specific χ/Q values for the EAB, LPZ outer boundary, MCR and present that information as part of the response to COL item 2.3-1.

Hazardous Material Releases

As stated in Section 2.2, the NuScale Power Plant certified design does not postulate any hazards from on-site sources or nearby industrial, transportation, or military facilities.

The COL applicant will provide discussion of site specific hazardous material releases as part of the response to COL item 2.3-1.

2.3.5 Long-Term Atmospheric Dispersion Estimates for Routine Releases

Site boundary annual average atmospheric dispersion factors (χ/Q values) and relative deposition factor (D/Q) values provided in Table 2.0-1 are used to calculate the site boundary release concentrations for comparison to the activity release limits in 10 CFR 20 as discussed in Section 11.3.

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~~Table 2.0-1 provides the annual average relative concentration (χ/Q values) at the site owner controlled area fence.~~

Annual average atmospheric dispersion factors (χ/Q values) and deposition factor (D/Q) values at the site boundary and at locations of interest are site-specific and are developed by the COL applicant as part of the response to COL Item 2.3-1.

2.3.6 References

2.3-1 National Oceanic and Atmospheric Administration Hydrometeorological Report Number 52, "Application of Probable Maximum Precipitation Estimates-United States East of the 105th Meridian," Washington DC, August 1982.

2.3-2 Electrical Power Research Institute, "Advanced Nuclear Technology: Advanced Light Water Reactor Utility Requirements Document," Revision 13, 2014.

2.3-3 NuScale Power LLC, Licensing Topical Report TR-0915-17565-P "Accident Source Term Methodology," Rev. 0, December 2015

2.3-4 American Society of Civil Engineers/Structural Engineering Institute, "Minimum Design Loads for Buildings and Other Structures," ASCE/SEI 7-05, Reston, VA, 2005.

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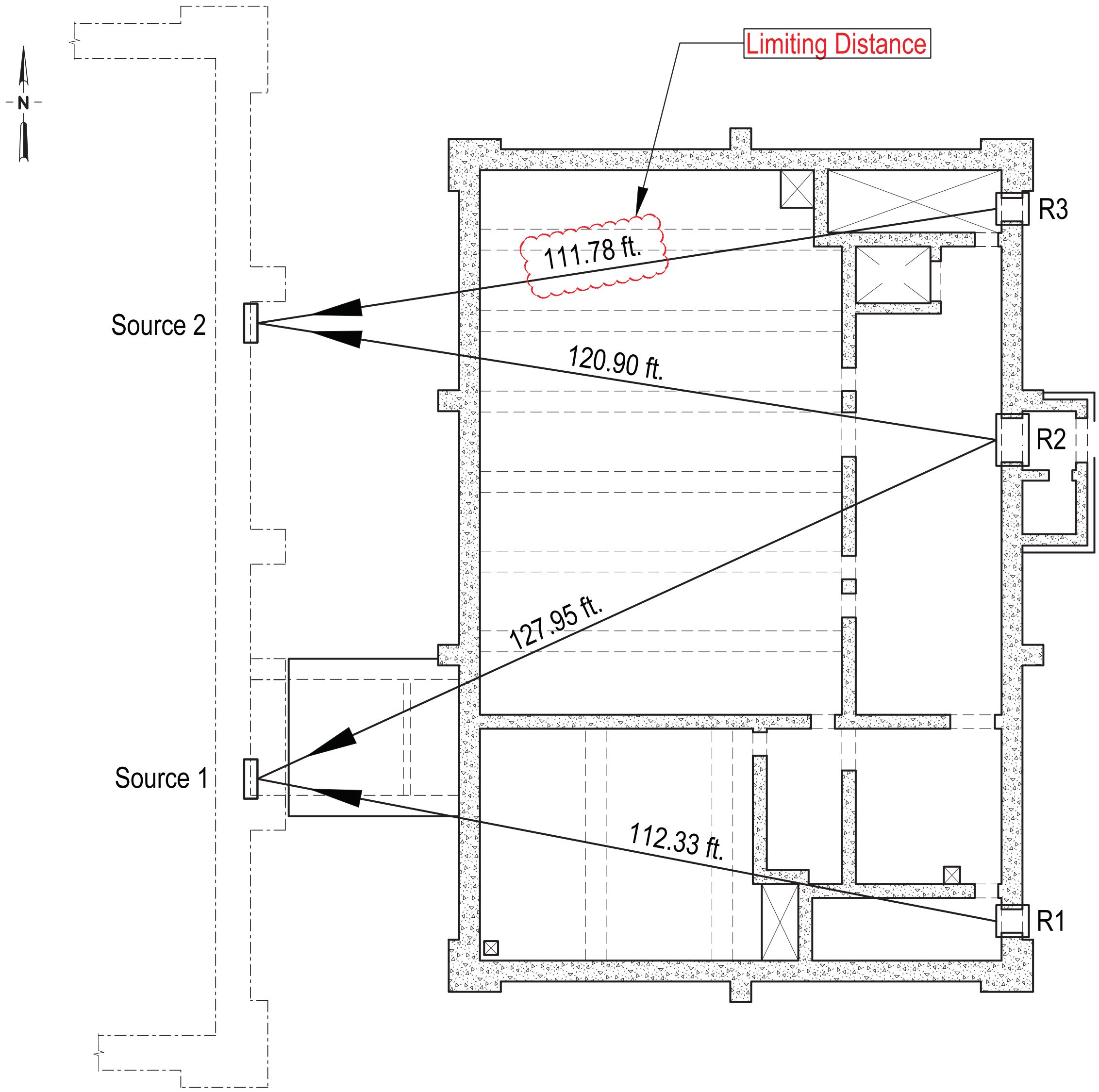
Figure 2.3-2: Source to Control Building Door Distances



Label	Description
R1	SE single personnel door (RWB)
R2	Double personnel door (RWB)
R3	NE single personnel door (RWB)
Source 1	SE single personnel door (RXB)
Source 2	NE single personnel Door (RXB)

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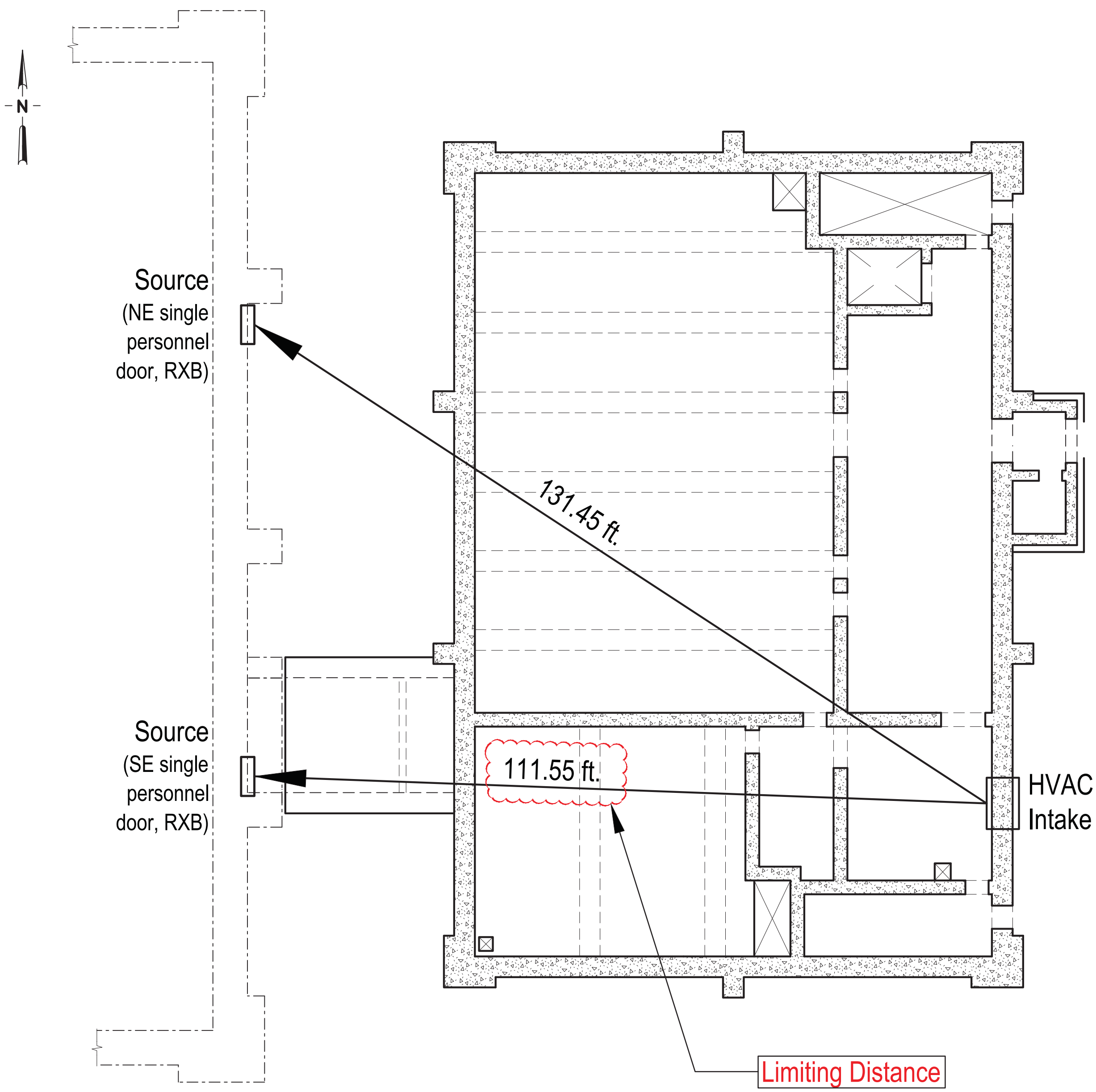
Figure 2.3-3: Source to Control Building HVAC Intake Distance



11.3.2.5 Site-Specific Cost-Benefit Analysis

Regulatory Guide 1.110 provides guidance for complying with 10 CFR 50, Appendix I, Section II, Paragraph D, to demonstrate that the addition of items of reasonably demonstrated technology is not favorable or cost-beneficial.

COL Item 11.3-1: A COL applicant that references the NuScale Power Plant design certification will perform a site-specific cost-benefit analysis.

11.3.2.6 Mobile or Temporary Equipment

The GRWS does not employ the use of mobile or temporary equipment in the design.

11.3.2.7 Seismic Design

The gaseous radioactive waste equipment and piping are classified in accordance with RG 1.143. The RWB seismic design is described in Section 3.7.2. The structures, systems, and component classifications for the GRWS components are listed in Table 3.2-1 and Table 11.3-2. The component activity contents are shown in Section 12.2.1.

11.3.3 Radioactive Effluent Releases

The GRWS processes and releases waste gas from normal reactor operations and AOOs to the RWBVS, and the waste gas is monitored and released to the environment through the RBVS exhaust stack. Section 9.4.2 provides additional information on the plant exhaust stack. Other normal gaseous discharge pathways include the condenser air removal system and secondary system steam leaks as illustrated in Figure 11.5-1. The discharge of gaseous effluents is tabulated by isotope, pathway, and annual released activity in Table 11.3-5.

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As described in Section 11.2.3, an alternate methodology to replace PWR-GALE was developed that uses first principles based calculations, combined with more recent nuclear industry experience. The calculation of gaseous effluent offsite dose consequences is consistent with methodologies presented in RG 1.112 and RG 1.109. A description of the methodology used to develop the primary and secondary coolant source terms is provided in Section 11.1. For normal effluents, the realistic coolant source terms are used and propagated through the plant systems. The major assumptions and inputs for the gaseous release methodology are listed in Table 11.3-4. From the component and airborne source terms, the normal gaseous effluent source term is determined and presented in Table 11.3-5. From the gaseous effluent source term, the offsite consequences are calculated using GASPAR II from the input values presented in Table 11.3-6. The atmospheric dispersion and deposition values presented in Table 11.3-6 were derived using NARCON and the assumptions presented in Table 11.3-12. The released gaseous radioactive effluent meets the concentration limits of 10 CFR 20.1302 and the dose limits of 10 CFR 50 Appendix I.

Gaseous Radioactive Waste System

RAI 02.03.04-1

Table 11.3-12: Assumptions for Routine Airborne Effluent Release Point Characteristics for Offsite Receptors

Parameter	Value
Release location	Plant exhaust stack
Release height	37.0 meters
Intake height	0.0 meters
Vent/stack exit velocity	0.0 meters/second
Vent/stack inside diameter	0.0 meters
Vent/stack exhaust orientation (vertical, horizontal, or other)	Not applicable
Restrictions to exhaust Air flow (e.g., rain caps)	Not applicable
Adjacent building height	0.0 meters
Adjacent building cross-sectional area	0.01 square meters

- The RADTRAD decay and daughter product 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) is the sum of the committed effective dose equivalent from inhalation and the deep dose equivalent from external exposure from tracked isotopes.
- RADTRAD does not include corrections for depletion of the effluent plume by deposition on the ground.
- 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.

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15.0.3.3.10 Post-Accident pH_T Code

The DBST methodology calculates the post-accident temperature-dependent pH. The pH_T code is used to calculate the extent of iodine re-evolution inside containment. During the postulated DBST event, additional acids and bases may enter the coolant and cause a change in pH. The expected overall pH of the coolant is modeled over a time period of 30 days. Section 15.0.2.4 provides a discussion of the NuScale pH_T program used to calculate post- accident temperature-dependent pH.

Details about the methodologies used for evaluating post-accident temperature-dependent pH in coolant water following a significant core damage event are presented in Reference 15.0-4. The results of implementing the methodology show that the post-accident temperature-dependent pH inside containment is between 6.0 and 7.0.

15.0.3.3.11 Iodine Re-evolution

The DBST methodology assumes a negligible amount of iodine re-evolution occurs between temperature-dependent pH values of 6.0 and 7.0 and 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. The treatment of iodine re-evolution is described in Reference 15.0-4.

15.0.3.3.12 Atmospheric Dispersion Factors (χ/Q), Breathing Rates, and Occupancy Factors

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Atmospheric dispersion factor (χ/Q) inputs to RADTRAD are derived as described in Reference 15.0-4 with assumptions shown in Table 15.0-20. Table 2.0-1 provides the accident release χ/Q values.

Control room and offsite breathing rate and control room occupancy factor inputs to RADTRAD, consistent with RG 1.183, are listed in Table 15.0-13.

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Table 15.0-20: Assumptions for Accident Airborne Effluent Release Point Characteristics for Offsite Receptors

Parameter	Value
Release location	Any point on Reactor Building or Turbine Generator Building wall
Release height	Ground level (0.0 meters)
Intake height	0.0 meters
Adjacent building cross-sectional area	Negligible (0.01 square meters)