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SMR, LLC		8002
Sponsoring Company		**Project No.**
HI-2241098	0	06 Jun 2025
Company Record Number	**Revision No.**	**Issue Date**
Report	Copyright	
Record Type	**Proprietary Classification**	
Nuclear	No	
Quality Class	**Export Control Applicability**	

Record Title:

SMR-300 Design Basis Accident Radiological Consequences Analysis Methodology Licensing Topical Report

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Revision Log

Revision	Description of Changes
0	Initial Issue.

Executive Summary

This topical report describes the methodology employed to analyze the radiological consequences of Design Basis Accidents (DBAs) for the SMR-300 design. The methodology incorporates the use of computational codes and justified assumptions to assess the Total Effective Dose Equivalent (TEDE) for receptors at the Exclusion Area Boundary (EAB), at the Low Population Zone (LPZ) outer boundary, in the Main Control Room (MCR), and in the Technical Support Center (TSC). The methodology can be used to demonstrate that the SMR-300 design is compliant with the United States Nuclear Regulatory Commission (NRC) regulatory requirements in 10 Code of Federal Regulations (CFR) 50.34(a)(1)(ii)(D) [1], § 50.34(b)(11) [1], and General Design Criterion (GDC) 19 in Appendix A to 10 CFR Part 50 [2].

The methodology is site-independent and adheres to NRC regulatory requirements and guidance with the following justified deviations from Regulatory Guide (RG) 1.183, Revision 1 [3]:

- Application of Assumption A-1.1 of RG 1.183 for iodine chemical forms in Maximum Hypothetical Accident (MHA) Loss of Coolant Accident (LOCA) analyses when containment flood-up pool pH ≥ 6.
- Application of Assumption H-4 of RG 1.183 for iodine chemical forms in Control Rod Ejection Accident (REA) analyses when containment flood-up pool pH ≥ 6.
- Application of Assumption B-2 of RG 1.183 for elemental iodine Decontamination Factor (DF) computation for Phase 1 release in Fuel Handling Accident (FHA) analyses when Spent Fuel Pool (SFP) water depth is greater than 23 feet.

Additional aspects of the methodology that adhere to NRC regulatory guidance, but may be considered unique, include:

- Determination of the “at-instant” core radionuclide inventory, consistent with Regulatory Position 3.1 of RG 1.183. The “at-instant” inventory represents radionuclide activity values at the instant of an accident, which is chosen to be bounding for the analyzed accident type.
- Application of the model in NUREG/CR-6189 [4] to determine the 10th percentile decontamination coefficient for natural aerosol deposition inside containment.
- Application of the equation in NUREG-0800, Section 6.5.2, Revision 4 [5] for the elemental iodine removal coefficient for natural wall deposition inside containment.
- Application of iodine spiking assumptions from RG 1.183 for secondary plant release path accidents with no fuel breach due to either cladding failure or fuel melt.

This topical report is independent of upstream analyses that provide inputs required to implement the methodology, such as thermal-hydraulic analysis to determine break flow due to Steam Generator Tube Rupture (SGTR), Locked Rotor and Rod Ejection Accident (REA) analysis to determine failed fuel fractions and mechanisms, and containment flood-up pool pH analysis.

Holtec requests NRC approval that the methodology described herein is technically acceptable and consistent with current regulations.

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1.0 INTRODUCTION

1.1 Purpose

The purpose of this topical report is to describe and justify a methodology to perform the radiological consequence analyses of Design Basis Accidents (DBAs) for the SMR-300 design, including the computational codes and assumptions used.

The methodology can be used to demonstrate that the SMR-300 design is compliant with the United States Nuclear Regulatory Commission (NRC) regulatory requirements in 10 Code of Federal Regulations (CFR) 50.34(a)(1)(ii)(D) [1], § 50.34(b)(11) [1], and General Design Criterion (GDC) 19 in Appendix A to 10 CFR Part 50 [2].

Holtec requests NRC approval that the methodology described herein is technically acceptable and consistent with current regulations.

1.2 Scope

The methodology can be used to determine the Total Effective Dose Equivalent (TEDE) for:

- An individual at the most limiting Exclusion Area Boundary (EAB) location.
- The most limiting receptor at the Low Population Zone (LPZ) outer boundary.
- The hypothetical maximum exposed individual in the Main Control Room (MCR).
- The hypothetical maximum exposed individual in the Technical Support Center (TSC).

The methodology is site-independent and adheres to NRC regulatory requirements and guidance with the following justified deviations from Regulatory Guide (RG) 1.183, Revision 1 [3]:

- Application of Assumption A-1.1 of RG 1.183 for iodine chemical forms in Maximum Hypothetical Accident (MHA) Loss of Coolant Accident (LOCA) analyses when containment flood-up pool[1] pH ≥ 6.
- Application of Assumption H-4 of RG 1.183 for iodine chemical forms in Control Rod Ejection Accident (REA) analyses when containment flood-up pool pH ≥ 6.
- Application of Assumption B-2 of RG 1.183 for elemental iodine decontamination factor (DF) computation for Phase 1 release in Fuel Handling Accident (FHA) analyses when Spent Fuel Pool (SFP) water depth is greater than 23 feet.

[1] The SMR-300 Reactor Pressure Vessel is located in the reactor cavity adjacent to the Spent Fuel Pool (SFP), both inside a dry containment. During a DBA in which coolant floods the containment, the reactor cavity and SFP are flooded to form a single pool, which is referred to as the “containment flood-up pool” in this report.

Additional aspects of the methodology that adhere to NRC regulatory guidance but may be considered unique include:

- Determination of the “at-instant” core radionuclide inventory, consistent with Regulatory Position 3.1 of RG 1.183. The “at-instant” inventory represents radionuclide activity values at the instant of an accident, which is chosen to be bounding for each analyzed accident.
- Application of the model in NUREG/CR-6189 [4] to determine the 10th percentile decontamination coefficient for natural aerosol deposition inside containment.
- Application of the equation in NUREG-0800, Section 6.5.2, Revision 4 [5] for the elemental iodine removal coefficient for natural wall deposition inside containment.
- Application of iodine spiking assumptions from RG 1.183 for secondary plant release path accidents with no fuel breach in the form of cladding failure or fuel melt.

This topical report is independent of upstream analyses that provide inputs required to implement the methodology. The following supporting analyses are beyond the scope of this report:

- System thermal-hydraulic analysis to determine break flow and its thermal-hydraulic properties due to Steam Generator Tube Rupture (SGTR).
- Locked Rotor and REA analyses to determine failed fuel fraction and failure mechanisms (cladding failure and/or fuel melt).
- pH analysis to show that the containment flood-up pool will have pH ≥ 6 for all normal and DBA conditions.

The methodology references Technical Specifications (TS). If final TS are not available at the time of analysis, preliminary TS values shall be used.

2.0 REGULATORY BASIS AND GUIDANCE

The radiological consequences analysis methodology detailed in this report is guided by the following regulatory requirements and guidance documents.

10 CFR 50.34 [1]: *Contents of Applications; Technical Information.*

Defines preliminary and final safety analysis report content requirements and acceptance criteria at EAB and LPZ for the radiological consequences analysis of DBAs.

Appendix A to 10 CFR Part 50 [2]: *General Design Criteria for Nuclear Power Plants*

GDC 19: *Control Room*

Defines regulatory requirements for radiation exposure limits in the control room during design basis accidents.

NUREG-0800: *Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition*

Section 6.5.2, Revision 4 [5]: *Containment Spray as a Fission Product Cleanup System*

Provides guidance for natural elemental iodine removal by wall deposition inside containment.

Section 15.0.3, Initial Issuance [6]: *Design Basis Accident Radiological Consequence Analyses for Advanced Light Water Reactors*

Provides guidance for acceptable criteria for assessing public and control room operator radiological exposures under accident conditions, ensuring that calculated doses do not exceed NRC regulatory requirements for advanced reactor designs.

RG 1.23, Revision 1 [7]: *Meteorological Monitoring Programs for Nuclear Power Plants*

Provides guidance on establishing meteorological monitoring programs at nuclear power plants to support safety analyses and emergency planning. It specifies requirements for accurate data collection of wind, temperature, and other parameters essential for atmospheric dispersion modeling and radiological impact assessments.

RG 1.183, Revision 1 [3]: *Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors*

Provides guidance for use of Alternative Source Terms in DBA dose consequence evaluations. It outlines acceptable methods and assumptions for calculating radiological doses to the public and plant personnel during accident scenarios.

RG 1.194, Revision 0 [8]: *Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants*

Provides guidance for calculation of atmospheric dispersion factors (χ/Q) to assess control room radiological habitability at nuclear power plants using ARCON. It also outlines acceptable assumptions and input parameters for the analyses.

RG 1.249, Revision 0 [9]: *Use of ARCON Methodology for Calculation of Accident-Related Offsite Atmospheric Dispersion Factors*

Provides guidance on use of the ARCON code to calculate offsite atmospheric dispersion factors (χ/Q) during nuclear plant accident scenarios. Additionally, it clarifies that ARCON96 and ARCON 2.0 use the same dispersion algorithms, differing only in their user interface.

NUREG/CR-5950 [10]: *Iodine Evolution and pH Control*

Provides the basis for elemental iodine chemical form fraction when the containment flood-up pool pH is maintained at or above 6.

NUREG/CR-6189 [4]: *A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments*

Provides the basis for an acceptable natural aerosol deposition model, which is integrated into RADTRAD as the “Powers’ Deposition Model”.

NUREG/CR-6331, Revision 1 [11]: *Atmospheric Relative Concentrations in Building Wakes*

Provides guidance and the technical basis for using the ARCON code for atmospheric dispersion analysis.

NUREG/CR-7220 [12]: *SNAP/RADTRAD 4.0: Description of Models and Methods*

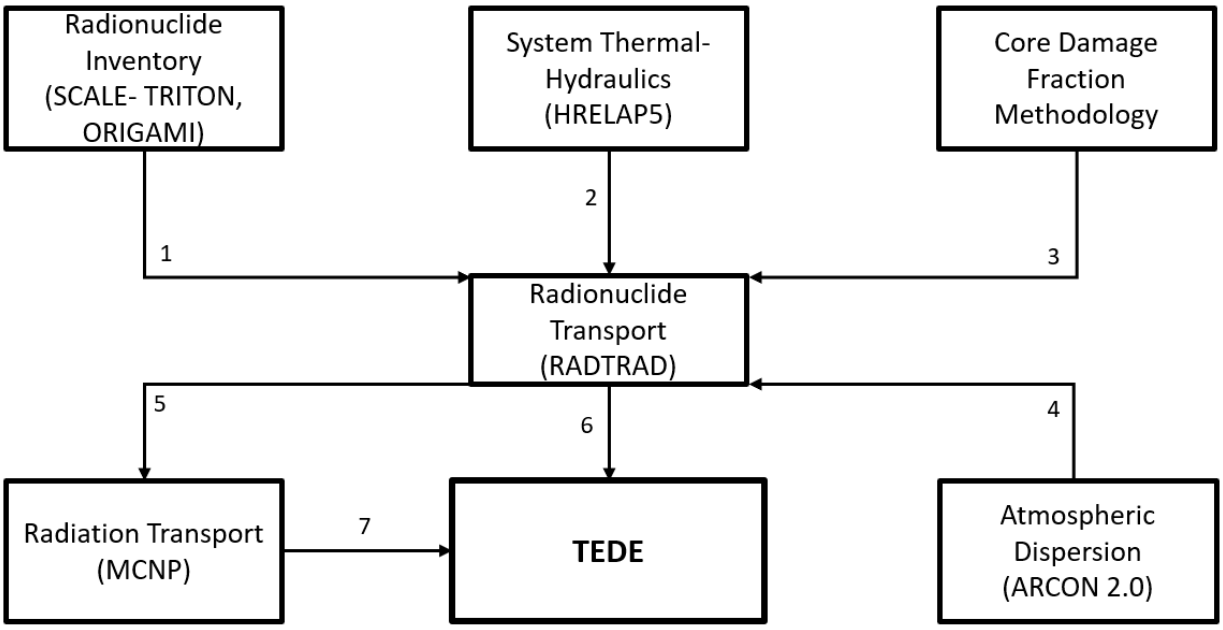
Provides guidance and the technical basis for using the RADTRAD code for radionuclide transport analysis.

NUREG-0933 Issue 197 [13]: *Iodine Spiking Phenomena*

Presents the methodology to determine the iodine release rate, which is used to calculate the design basis primary coolant inventory.

3.0 CODE DESCRIPTIONS AND ANALYSES FLOWCHART

Figure 3-1 shows a flowchart for the DBA radiological consequences analysis methodology. Code descriptions can be found in the subsections that follow.



1	Inventories (Core, Primary and Secondary Coolants)
2	Primary and Secondary Coolant Thermal-Hydraulic Conditions
3	Failed Fuel Fractions
4	Atmospheric Dispersion Factors
5	Containment, Plume, and Filters Inventory
6	Immersion and Inhalation Dose
7	Radiation Shine Dose

Figure 3-1 DBA Radiological Consequences Analysis Methodology Flowchart

3.1 SCALE 6.2.1: TRITON and ORIGAMI Modules

For radionuclide inventory calculations, the SCALE 6.2.1 (SCALE) [14] simulation suite is used. SCALE is a multi-application code system with tools for reactor physics, criticality safety, radiation shielding, and spent fuel characterization for a range of nuclear systems. SCALE is developed and maintained by Oak Ridge National Laboratory.

The radiological consequences analyses use two of the SCALE computational modules:

1) TRITON
    - The TRITON computer code is a multipurpose SCALE control module for transport, depletion, and sensitivity and uncertainty analysis. TRITON can be used to provide automated, problem-dependent cross-section processing followed by multigroup transport calculations for one-, two-, and three-dimensional (1D, 2D, and 3D) configurations.
2) ORIGAMI
    - SCALE includes the ORIGAMI (ORIGEN Assembly Isotopics) tool to compute detailed isotopic compositions for light water reactor assemblies with UO_2 fuel.

TRITON is used as a control module for transport and depletion calculations. It is used to prepare problem- and exposure-dependent multigroup cross-sections. TRITON uses these cross-sections to perform 2D deterministic transport calculations. TRITON then initiates ORIGAMI for depletion calculations. ORIGAMI performs ORIGEN burnup calculations for each of the specified power regions to obtain the spatial distribution of isotopes in the burned fuel. The process repeats until the simulation reaches its end time.

3.2 HRELAP5

System thermal-hydraulics analyses are performed using the HRELAP5 computer code. HRELAP5 is a Holtec proprietary code based on the RELAP5-3D code developed by the Idaho National Laboratory.

The code models the coupled behavior of the reactor coolant system and the core for DBAs and Anticipated Operational Occurrences. A generic modeling approach is used that permits simulation of a variety of thermal-hydraulic systems. Control system logic and auxiliary system components are included to model plant controls and appropriate system responses.

3.3 ARCON 2.0

ARCON 2.0 [15] is used to calculate onsite and offsite atmospheric dispersion factors (χ/Q). ARCON 2.0 is implemented in a manner that is consistent with the guidance provided in RG 1.194 and RG 1.249. As discussed in RG 1.249, ARCON 2.0 has the same underlying dispersion algorithms as ARCON96 and solely provides an updated user interface. ARCON implements a building wake dispersion algorithm and a straight-line Gaussian diffusion model. Here, ARCON refers to both the ARCON96 and ARCON 2.0 versions of the code.

3.4 RADTRAD 5.0.3

RADTRAD 5.0.3 (RADTRAD) [16] is used to calculate radionuclide transport, removal of radionuclides, and TEDE at selected receptor sites (EAB, LPZ outer boundary, MCR, and TSC). RADTRAD inputs include radionuclide inventories, release fractions, release timing, and atmospheric dispersion factors.

Radionuclide transport is based on the material flow between buildings, from buildings to the environment, or into the control room through filters, piping, or other pathways. An accounting of the amount of radioactive material retained in these pathways is maintained. Decay and in-growth of daughters can be calculated over time as material is transported.

3.5 MCNP 6.2

MCNP 6.2 (MCNP) [17] is utilized for evaluating shine radiological exposures or doses to operators within the MCR and/or TSC following a radiological release event. Direct shine, sky-shine, and shine from filters are evaluated.

MCNP is a general-purpose tool used for neutron, photon, electron, or coupled neutron, photon, and electron transport. MCNP treats an arbitrary 3D configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. The code is well-suited for performing fixed source calculations as required for the radiation shine analysis.

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

4.0 METHODOLOGY

This section describes the site-independent, conservative methodology for performing radiological consequences analyses of DBAs for the SMR-300 design. The methodology is divided into subsections listed below:

- Accident Release Durations and Dose Acceptance Criteria
- Radionuclide Inventory
- Source Terms
- Radionuclide Transport
- Dose Consequences

Subsection 4.1 “Accident Release Durations and Dose Acceptance Criteria” describes the selection of the analysis duration and TEDE acceptance criteria for each accident to be analyzed. The subsection lists credible DBAs based on the SMR-300 design and guidance provided by RG 1.183.

Subsection 4.2 "Radionuclide Inventory" describes the calculation of the core and coolant radionuclide inventories available for release.

Subsection 4.3 "Source Terms" describes the calculation of source terms for various accident scenarios based on the radionuclide inventories and release characteristics.

Subsection 4.4 "Radionuclide Transport" describes the radionuclide transport analysis based on the source terms and radionuclide transport phenomena. This subsection discusses factors such as SFP decontamination, natural aerosol deposition, elemental iodine wall deposition, containment leakage, secondary plant release, atmospheric dispersion, MCR habitability systems, and TSC habitability systems.

Subsection 4.5 “Dose Consequences” describes the calculation of TEDE. This subsection describes conservative calculation of the immersion and inhalation doses received by receptors at the EAB, at the LPZ outer boundary, in the MCR, and in the TSC. This subsection also describes the calculation of radiation shine dose contribution to TEDE.

4.1 Accident Release Durations and Dose Acceptance Criteria

Table 4.1 summarizes the fission product release durations considered for the DBA radiological consequences analysis, which are consistent with the referenced regulatory requirements and guidance.

Table 4.1 Analysis Release Duration

Receptor Location	Time Duration[2,3]	References
EAB	Worst 2-hour window in 720 hours (30 days)	§ 50.34 and RG 1.183[4]
LPZ	720 hours (30 days)	§ 50.34 and RG 1.183[5]
MCR	720 hours (30 days)	GDC 19 and RG 1.183[6]
TSC	720 hours (30 days)	NUREG-0800, Section 15.0.3 and RG 1.183[7]

[2] Consistent with Table 7 of RG 1.183, accident durations for SGTR, Main Steam Line Break, Locked Rotor Accident, and REA (containment and secondary plant release paths) are conservatively bounded by 30 days or the time until shutdown cooling is in operation and releases from the steam generator have been terminated.

[3] The accident duration for “Failure of Small Lines Carrying Primary Coolant Outside Containment” is conservatively bounded by 30 days or the time until leakage stops when the reactor is shut down and depressurized.

[4] § 50.34(a)(1)(ii)(D)(1) and Regulatory Position 4.1.e of RG 1.183 provide the basis for the analysis window. Table 7 of RG 1.183 provides the basis for accident duration/analysis release duration.

[5] § 50.34(a)(1)(ii)(D)(2) and Regulatory Position 4.1.f of RG 1.183 provide the basis for analysis over the accident duration. Table 7 of RG 1.183 provides the basis for accident duration/analysis release duration.

[6] GDC 19 in Appendix A to 10 CFR Part 50 provides the basis for analysis over the accident duration. Footnote 2 to Table 7 of RG 1.183 provides the basis for accident duration/analysis release duration.

[7] Acceptance Criteria 3 of NUREG-0800, Section 15.0.3 provides the basis for analysis over the accident duration. Footnote 2 to Table 7 of RG 1.183 provides the basis for accident duration/analysis release duration.

Table 4.2 lists the TEDE acceptance criteria over the analyzed duration, which satisfy the requirements in § 50.34(a)(1)(ii)(D) and GDC 19, and is consistent with the guidance in Table 7 of RG 1.183. The event "Failure of Small Lines Carrying Primary Coolant Outside Containment" is discussed in NUREG-0800, Sections 15.0.3 [6] and 15.6.2 [18], but it is not addressed in RG 1.183. It is analyzed due to its potential for credible radionuclide release into the environment.

Table 4.2 TEDE Acceptance Criteria

Accident	**EAB and LPZ Dose Criteria (TEDE)**	**MCR and TSC Dose Criteria (TEDE)**
MHA LOCA	25 rem	5 rem
SGTR		
Pre-Accident Spike	25 rem	5 rem
Concurrent Spike	2.5 rem	5 rem
Main Steam Line Break		
Pre-Accident Spike	25 rem	5 rem
Concurrent Spike	2.5 rem	5 rem
Locked Rotor	2.5 rem	5 rem
REA		
Containment Release Path	6.3 rem	5 rem
Secondary Plant Release Path	6.3 rem	5 rem
FHA	6.3 rem	5 rem
Failure of Small Lines Carrying Primary Coolant Outside Containment		
Pre-Accident Spike	25 rem	5 rem
Concurrent Spike	2.5 rem	5 rem

4.2 Radionuclide Inventory

The methodology for calculating core, primary coolant, and secondary coolant radionuclide inventories is described in the following subsections.

The SMR-300 SFP is located inside containment. The radionuclide inventory of the spent fuel stored in the SFP and the SFP coolant is not considered in this methodology. The design of the SMR-300 precludes damage to fuel in the SFP as a result of a DBA. The source term used for the FHA is discussed in subsection 4.3.5.

4.2.1 Core Radionuclide Inventory

The core radionuclide inventory is conservatively calculated using SCALE; the calculation is performed in a manner consistent with the guidance in Regulatory Position 3.1 of RG 1.183. The inventory is calculated for the elements listed in Table 6 of RG 1.183.

The inventory of a single assembly is calculated based on the fuel assembly geometry, assembly Radial Peaking Factor (RPF), rated power with uncertainty, assembly-average exposure, and U-235 enrichment. To determine the core inventory for a specific accident scenario, the single assembly inventory is multiplied by the total number of assemblies in the core and the fraction of the failed fuel. For the MHA LOCA, full core melt is assumed.

Using the approach outlined above, three “at-instant” core inventories are calculated. The term “at-instant” reflects that activity values are taken at the instant of an accident. The instant is chosen to be bounding for the analyzed accident type.

The three “at-instant” core inventories calculated are:

1) Radionuclide Inventory with RPF
   - This inventory is used for scenarios involving partial core damage.
2) Radionuclide Inventory without RPF
   - This inventory is used for scenarios assuming full-core damage. The assembly RPF is not applied, consistent with Regulatory Position 3.1 of RG 1.183.
3) Radionuclide Inventory with RPF and Conservative Cooling Time After End of Equilibrium Cycle
   - This inventory is applied for FHA analysis.

For all accidents except the FHA, the burnup used to calculate the radionuclide inventory bounds the highest burnup that may occur during operation, which is achieved at the end of a fuel cycle. For all accidents except the FHA, the “at-instant” core inventory therefore reflects an assumption that the accident occurs at the end of a fuel cycle. This assumption is conservative because radionuclide buildup increases with burnup, leading to a higher predicted TEDE.

For the FHA, the accident is assumed to occur after a conservative cooling period following reactor shutdown at the end of a fuel cycle. The selected cooling period is less than or equal to the TS minimum time before commencing refueling operations following a reactor shutdown (the time between reactor shutdown and the engagement of any assembly for refueling operations). Radioactive decay is credited for the duration of the cooling period.

4.2.2 Coolant Radionuclide Inventory

The methodology used to calculate the primary and secondary coolant inventories is described in the following subsections.

4.2.2.1 Primary Coolant Inventory

The principal mechanism for the presence of radioactive materials in the primary coolant is fission product leakage into the coolant. The primary coolant inventory is determined based on the reactor core inventory with RPF. The extent of fission product leakage is conservatively

determined based on the failed fuel fraction, or, if no fuel breach occurs, from the maximum fuel defect level allowed during operation as specified in the TS. Escape rate coefficients from fuel to coolant are provided in Table 4.3 [19]. These coefficients are applicable to the SMR-300 design; the SMR-300 utilizes typical Pressurized Water Reactor (PWR) fuel and has a linear heat generation rate and core average temperature comparable to typical large PWRs.

During normal operations, the primary coolant is continuously purified by filters and demineralizers in the Chemical and Volume Control System. The concentration of fission products in the primary coolant system is reduced through purification, leakage to the secondary coolant, and addition of makeup water. In calculating primary coolant source terms, credit is taken for purification and makeup water additions, but not for leakage to the secondary coolant.

Table 4.3 Fission Product Escape Rate Coefficients

Isotope[8]	Escape Rate Coefficient (s^{-1})
Xe and Kr isotopes	6.5×10^{-8}
I, Br, Rb and Cs isotopes	1.3×10^{-8}
Mo isotopes	2.0×10^{-9}
Te isotopes	1.0×10^{-9}
Sr and Ba isotopes	1.0×10^{-11}
Y, Zr, Nb, Tc, Ru, La, and Ce isotopes	1.6×10^{-12}

4.2.2.2 Secondary Coolant Inventory

Leakage of primary coolant into the secondary coolant in the steam generator is the sole source of radioactivity in the secondary coolant system. Purification of secondary coolant by deaeration in the condenser and by demineralization in the condensate polisher is credited.

4.3 Source Terms

Source terms refer to the amount, type, and behavior of radioactive material that could be released from a nuclear facility into the environment. The source terms are determined using the following information:

1) Radionuclide inventories
2) Release fractions
3) Release timings
4) Chemical form
5) Iodine spiking (for primary coolant source terms when there is no fuel breach)

[8] The isotopes considered for the primary coolant inventory calculations are only those that may reside in the fuel rod plenum and gap consistent with Table 4 and Table 6 of RG 1.183. The isotopes to consider are those of Xe, Kr, I, Br, Rb, and Cs.

4.3.1 MHA LOCA Source Terms

The source terms calculation methodology in this subsection is applied for MHA LOCAs. The source terms are developed to comply with Footnote 3 to § 50.34, which assumes the accident will result in substantial meltdown of the core.

4.3.1.1 Release Fractions and Timings

MHA LOCA release fractions are obtained from Table 2 of RG 1.183. MHA LOCA release timings are obtained from the PWR release timings in Table 5 of RG 1.183.

4.3.1.2 Iodine Chemical Form

Consistent with Regulatory Position 3.5 of RG 1.183, the chemical form fractions of iodine considered in the MHA LOCA analysis are as follows:

- 95% Cesium Iodide
- 4.85% Elemental Iodine
- 0.15% Organic Iodide

Assumption A-1.1 of RG 1.183 recommends these chemical form fractions for the MHA LOCA analysis if containment flood-up pool pH is 7 or greater. However, these chemical form fractions remain applicable if containment flood-up pool pH is 6 or greater because the iodine evolution to elemental iodine (I_2) is significantly less than 4.85% at a pH of 6, as shown in Figure 4-1 (reproduced from NUREG/CR-5950, Figure 3.1).

Figure 4-1 and the associated model for I_2 evolution as a function of pH presented in NUREG/CR-5950 is applicable to the SMR-300 design. While the SMR-300 containment flood-up pool pH analysis is outside the scope of this report, the methodology will be consistent with NUREG/CR-5950, which models chemistry based on radiolytic acid generation – primarily from nitric acid (HNO_3), hydrochloric acid (HCl), and hydrogen iodide (HI) – and buffering by added bases such as sodium tetraborate. The iodine concentration will be less than 10^{-4} gram-atoms of iodine per liter (g-at. I/L) for the SMR-300 for all normal and DBA scenarios.

Figure 4-1 relates I_2 speciation to pH at 77°F. The rate constants utilized in the generation of Figure 4-1 were therefore calculated at 77°F. The rate constants are calculated as a function of temperature; the function is expected to produce reasonable predictions at temperatures close to 77°F. The model predicts lower I_2 fractions at higher temperatures. Furthermore, Appendix C to NUREG/CR-5950 refers to experimental results suggesting that model predictions exceed measured I_2 fractions at higher temperature. Application of Figure 4-1 to relatively high-temperature accident conditions over-predicts speciation of I_2 in the containment. Therefore, 4.85% elemental iodine is an appropriate chemical form fraction for the SMR-300 accident analysis.

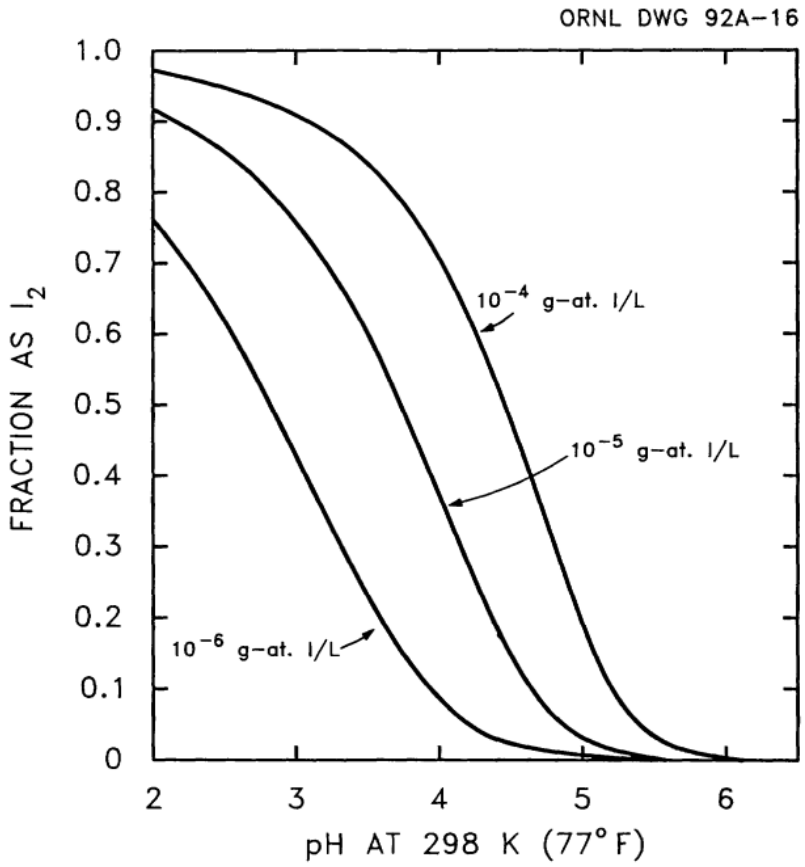


Figure 4-1 Model Calculations of Fraction as I_2 vs pH

4.3.2 REA Source Terms for Containment Release Path

The source terms calculation methodology in this subsection is applied for the limiting REA containment release path analysis if fuel breach is postulated. If fuel breach is not postulated for the limiting event, the radiological analysis is not performed, as permitted by Assumption H-2 of RG 1.183.

4.3.2.1 Release Fractions and Timings

Consistent with Assumption H-1 of RG 1.183, all of the noble gases and half of the iodines are assumed to be released from the fuel rod plenum and gap. The release fractions are obtained from Table 4 of RG 1.183. To be conservative, the release is assumed to be instantaneous.

4.3.2.2 Iodine Chemical Form

Consistent with Regulatory Position 3.5 of RG 1.183, the chemical form fractions of iodine considered in the REA containment release path analysis are as follows:

- 95% Cesium Iodide
- 4.85% Elemental Iodine
- 0.15% Organic Iodide

Assumption H-4 of RG 1.183 recommends these chemical form fractions for the REA analysis if containment flood-up pool pH is 7 or greater. However, as justified in subsection 4.3.1.2, these chemical form fractions remain applicable if containment flood-up pool pH is 6 or greater.

4.3.3 REA Source Terms for Secondary Plant Release Path

The source terms calculation methodology in this subsection is applied for the limiting REA secondary plant release path analysis if fuel breach is postulated. If fuel breach is not postulated for the limiting event, the radiological analysis is not performed, as permitted by Assumption H-2 of RG 1.183.

4.3.3.1 Release Fractions and Timings

Consistent with Assumption H-1 of RG 1.183, all of the noble gases and half of the iodines are assumed to be released from the fuel rod plenum and gap. The release fractions are obtained from Table 4 of RG 1.183. To be conservative, the release is assumed to be instantaneous.

4.3.3.2 Iodine Chemical Form

Consistent with Assumption H-5 of RG 1.183, the chemical form fractions of iodine releases from the steam generator to the environment that are considered for the limiting REA secondary plant release path analysis are as follows:

- 97% Elemental Iodine
- 3% Organic Iodide

4.3.4 Locked Rotor Accident Source Terms

The source terms calculation methodology in this subsection is applied for Locked Rotor Accidents if fuel breach is anticipated. If fuel breach is not postulated for the limiting event, the radiological analysis is not performed, as permitted by Assumption G-2 of RG 1.183.

4.3.4.1 Release Fractions and Timings

For the Locked Rotor Accident, fuel cladding failure is the only failure mechanism considered. The SMR-300 is designed to preclude fuel melt resulting from this event. Consequently, only isotopes of noble gases, halogens, and alkali metals (radionuclides present in the fuel rod plenum and gap) are considered for release. The release fractions are obtained from Table 4 of RG 1.183. To be conservative, the release is assumed to be instantaneous.

4.3.4.2 Iodine Chemical Form

Consistent with Assumption G-4 of RG 1.183, the chemical form fractions of iodine releases from the steam generator to the environment that are considered for the limiting Locked Rotor Accident analysis are as follows:

- 97% Elemental Iodine
- 3% Organic Iodide

4.3.5 FHA Source Terms

The source terms calculation methodology in this subsection is applied for FHAs. During fuel handling, the reactor cavity (containing the Reactor Pressure Vessel) and the adjacent SFP (inside containment) are flooded to an equivalent level. Each assembly is transferred one at a time from the Reactor Pressure Vessel to the SFP. The SMR-300 fuel handling bridge crane is single failure proof and designed such that an assembly cannot be inadvertently released by the crane. This ensures that an assembly cannot be dropped resulting in damage to another assembly. Further controls, including the fuel handling bridge crane overload cutout, are in place to ensure that fuel assemblies do not damage adjacent assemblies when being lifted by the crane. Therefore, assuming a single assembly is damaged with all its fuel rods failing is conservative. Hence, the radionuclide inventory for a single fuel assembly is used as an input for evaluation of the FHA.

4.3.5.1 Release Fractions and Timings

Consistent with Assumption B-1.2 of RG 1.183, all radionuclides in the fuel rod plenum and gap are assumed to be released from the breached fuel. The release fractions are obtained from Table 4 of RG 1.183. To be conservative, the release is assumed to be instantaneous.

4.3.5.2 Iodine Chemical Form

Consistent with Regulatory Position 3.5 and Assumption B-1.3 of RG 1.183, the chemical form fractions of iodine considered in the FHA analysis are as follows:

- 95% Cesium Iodide
- 4.85% Elemental Iodine
- 0.15% Organic Iodide

4.3.6 Primary Coolant Source Terms for DBAs with No Fuel Breach

The source terms calculation methodology in this subsection is applied for DBAs with no fuel breach in the form of cladding failure or fuel melt. Consistent with RG 1.183, if no fuel breach is postulated for the accident scenario, the activity released should be the maximum coolant activity allowed by the TS and consider both pre-accident and concurrent iodine spiking.

4.3.6.1 Pre-Accident Spike

The pre-accident spike refers to an increase in primary coolant activity that occurs immediately before the onset of an accident. It accounts for elevated radionuclide concentrations due to

potential cladding defects or operational conditions, which are then released during an event. Consistent with Assumptions E-2.1 and F-2.1 of RG 1.183, the primary coolant iodine activity is assumed to increase to the maximum value permitted at full-power operations by the TS.

Primary coolant iodine content may be expressed on an isotope-by-isotope basis or as Dose Equivalent (DE) I-131. DE I-131 represents the concentration of I-131 that alone would produce the same committed effective dose equivalent as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 present. The Dose Conversion Factors (DCF) used to convert between isotopic concentrations and DE I-131 are obtained from Table 2.1 of Federal Guidance Report (FGR) 11 [20].

4.3.6.2 Concurrent Spike

The concurrent spike describes a sharp rise in primary coolant activity that happens simultaneously with an accident, typically due to fuel damage or sudden releases caused by the initiating event. The increase in primary coolant iodine concentration is estimated using a spiking model, which assumes that the iodine release rate corresponding to the equilibrium concentration specified in the TS is amplified by a spiking factor.

The iodine release rate (sometimes referred to as the appearance rate, fission product release rate, or production rate) from the fuel rods to the primary coolant (expressed in curies per unit time) is used to determine the concurrent iodine spike. NUREG-0933 Issue 197 presents the methodology to determine the iodine release rate, which is used to calculate the design basis primary coolant inventory used in this analysis.

An iodine spike can be initiated by a power or pressure transient. Concurrent iodine spike activity is calculated following the process described below.

1) The iodine appearance rate is calculated based on the equilibrium between its production and removal rates. This can be expressed by the following equation:

$$\boldsymbol{R_0 = A_0\lambda_t = A_0(\lambda + \lambda_U)} \qquad \textbf{Equation 4-1}$$

where,

A_0= Equilibrium activity in the coolant, Ci

R_0 = Equilibrium release rate from the fuel, Ci/hr

λ_t = Total removal rate, s^{-1}

λ = Removal rate due to radioactive decay, s^{-1}

λ_U = Removal coefficient for purification (sum of removal rates due to demineralization and dilution), s^{-1}

2) The resulting appearance rate is multiplied by a spiking factor and spike duration to determine activity in the primary coolant due to a concurrent spike. Consistent with Assumptions E-2.2 and F-2.2 of RG 1.183, the spike duration is set to 8 hours, and the spiking factor is conservatively chosen as 500. Notably, Assumption E-2.2 prescribes a spiking factor of 335, which is bounded by the selected value of 500.

4.3.6.3 Release Fractions and Timings

The main contributor to primary coolant activity is fuel rods which have defects; for fuel rods with defects, radionuclides which are in the fuel rod plenum and gap are available for release. Only noble gases, halogens and alkali metals isotopes are considered in the analysis. Release fractions are obtained from Table 4 of RG 1.183. To be conservative, the release is assumed to be instantaneous.

4.3.6.4 Iodine Chemical Form

Consistent with Assumptions E-5 and F-5 of RG 1.183, the chemical form fractions of iodine considered in the analysis for DBAs with no fuel breach are as follows:

- 97% Elemental Iodine
- 3% Organic Iodide

4.3.7 Secondary Coolant Source Terms

The secondary coolant source terms have characteristics as stated in this subsection.

4.3.7.1 Release Fractions and Timings

Release fractions from fuel are not relevant to secondary coolant source term calculations as the coolant is not in contact with the fuel. Secondary coolant inventory, as determined in subsection 4.2.2.2, is assumed to be released to the environment via steaming without any mitigation.

4.3.7.2 Iodine Chemical Form

Consistent with Assumptions E-5 and F-5 of RG 1.183, the chemical form fractions of iodine considered for all DBAs with secondary plant release path are as follows:

- 97% Elemental Iodine
- 3% Organic Iodide

4.4 Radionuclide Transport

Once the core and coolant source terms are established, the transport of radionuclides is conservatively analyzed. This analysis includes evaluation of radionuclide transport within the containment, dispersion into the atmosphere, transmission through the MCR and TSC habitability systems, and absorption by receptors at various locations. The phenomena considered in the radionuclide transport analysis, along with the methodologies used to incorporate them, are described below.

4.4.1 Spent Fuel Pool Decontamination

When the most limiting FHA occurs, radioactive materials from the fuel rod plenum and gap are released upon the impact of the fuel assembly on the SFP floor. A filtered pathway is modeled to simulate the retention of radioactive materials as they rise through the SFP water column.

Consistent with the assumptions in Appendix B of RG 1.183, fission product retention in the SFP water column is credited. The modeling of retention varies by nuclide as described below:

- The retention of noble gases in the water in the SFP is negligible (i.e., the DF is 1).
- Particulate radionuclides are assumed to be retained by the water in the SFP (i.e., the DF is infinite).
- For elemental isotopes, the decontamination happens in two phases:
- Phase 1 is the instantaneous release from the rising bubbles (from start of accident to 2 hours). Phase 1 DF is conservatively calculated using Assumption B-2 of RG 1.183. For SFP depths greater than 23 feet, this is a conservative assumption as a deeper pool provides more scrubbing, resulting in larger DF.

$$DF = 81.046 \cdot e^{0.305\left(\frac{t}{d}\right)} \quad \textbf{Equation 4-2}$$

$$t = 9.2261 \cdot e^{-6 \cdot 10^{-4} \cdot x} \quad \textbf{Equation 4-3}$$

$$d = -0.0002 \cdot x + 1.0009 \quad \textbf{Equation 4-4}$$

where,

t = Bubble rise time $[s]$

x = Pin pressure $[psig]$

d = Bubble diameter $[cm]$

- Phase 2 is the protracted release due to re-evolution as elemental iodine (from 2 hours to 30 days). Since all radionuclides are assumed to be released instantaneously at the onset of the accident, no Phase 2 decontamination is credited.

The leakage filter efficiency due to the SFP water column is determined using the following equation:

$$\eta = 1 - \frac{1}{DF} \quad \textbf{Equation 4-5}$$

The containment leakage flow rate to the environment is conservatively assumed to be sufficiently large to simulate instantaneous release of radioactive materials to the atmosphere after they reach the surface of the SFP water column.

4.4.2 Natural Aerosol Deposition

Consistent with Assumption A-2.2 of RG 1.183, natural aerosol deposition is credited in radiological analyses for all accidents with containment path releases. The modeling of natural aerosol deposition is performed using the method outlined in NUREG/CR-6189. This method is integrated into RADTRAD as the “Powers’ Deposition Model” and is used to determine the 10th percentile aerosol decontamination coefficient, as endorsed by Assumption A-2.2 of RG 1.183.

4.4.3 Elemental Iodine Removal

Consistent with Assumption A-2.2 of RG 1.183, iodine removal via iodine gas interaction with free surfaces inside containment is credited in radiological analyses for all accidents with containment path releases.

The elemental iodine removal coefficient is determined using the equation below from NUREG-0800, Section 6.5.2. Containment building free volume and wetted surface area are determined conservatively.

$$\lambda_W = \frac{K_W A}{V} \qquad \textbf{Equation 4-6}$$

where,

λ_W= Elemental iodine removal coefficient by wall deposition $[hr^{-1}]$

V= Containment building free volume $[m^3]$

A = Wetted surface area $[m^2]$

K_W = 4.9 $[\frac{m}{hr}]$ (bounding mass-transfer coefficient per NUREG-0800, Section 6.5.2)

4.4.4 Containment Leakage

Consistent with Assumption A-2.7 of RG 1.183, radionuclides available for release are assumed to leak to the atmosphere for all accidents with containment path releases. This leakage is assumed to occur at the peak pressure TS leakage rate. The leakage is assumed to reduce to 50% of the TS leakage rate after 24 hours.

For FHA, the containment air lock is assumed to be open during fuel handling. Therefore, radioisotopes exposed to the containment environment are assumed to be released to the environment instantaneously.

4.4.5 Secondary Plant Release Path Accident Sources

For accidents involving secondary plant release paths, three sources of radioactive material release to the environment are considered:

1) Primary-to-secondary normal leakage
   - Considered for all accidents involving secondary plant release path
2) Primary-to-secondary break flow due to SGTR
   - Considered only for SGTR accident
3) Steaming of steam generator bulk water
   - Considered for all accidents involving secondary plant release path

Unlike the U-tube steam generator design that is assumed in RG 1.183, the SMR-300 design incorporates a Once-Through Steam Generator (OTSG), with the upper portion of the tubes located in the steam space. For conservatism, primary-to-secondary normal leakage and break flow due to SGTR in the OTSG are assumed to occur in the steam space and completely flash

to vapor. This leakage and break flow are assumed to be released directly to the environment, with no credit taken for partial flashing or scrubbing.

The normal primary-to-secondary leakage rate in the steam generator is assumed to be at the Limiting Condition of Operation specified in the TS.

The break flow rate associated with a SGTR is determined using conservative HRELAP5 calculations to maximize the leakage rate.

Credit is applied for the decay of radionuclides, including the formation of decay daughters, until their release to the environment.

4.4.6 Atmospheric Dispersion

In the event of an accident, radionuclides may be released to the environment from containment leakage or secondary plant release paths. These radionuclides form a plume that could travel to receptor sites at the EAB, at the LPZ outer boundary, in the MCR, and in the TSC. As the plume moves, it disperses, leading to a reduction in radionuclide concentration over distance. To quantify this dispersion and assess the potential exposure for a receptor at various locations, a site-specific atmospheric dispersion analysis is conducted.

Regulatory Position 5.3 of RG 1.183 prescribes the application of RG 1.194 and RG 1.249 for calculating atmospheric dispersion factors (χ/Q) if source-to-receptor distance is less than 1200 meters (m). Procedures consistent with RG 1.249 are used to support application of the modeling methodology in RG 1.194 for EAB and LPZ χ/Q calculations. The NRC-sponsored computer code ARCON 2.0 is used for the analysis and is endorsed by RG 1.249. As discussed in RG 1.249, ARCON 2.0 uses the same underlying dispersion algorithms from ARCON96, which is endorsed by RG 1.194; the only difference between ARCON 2.0 and ARCON96 is the user interface. Therefore, ARCON 2.0 and ARCON96 are equally applicable to the calculations.

4.4.6.1 Overview

The flowchart of the atmospheric dispersion analysis is shown in Figure 4-2. The steps are further described in the subsections below.

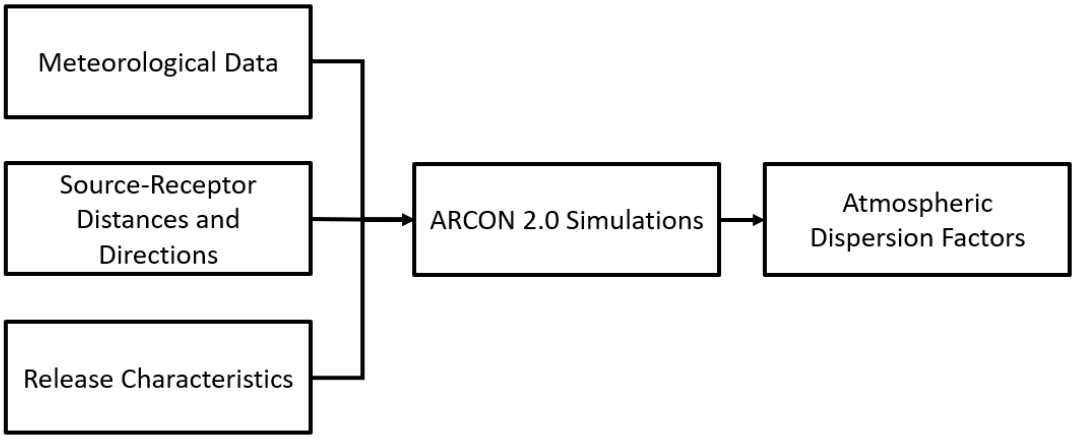


Figure 4-2 Atmospheric Dispersion Analysis Flowchart

4.4.6.2 Meteorological Data

The methodology described herein is site-independent; therefore, when applied to a particular site, hourly meteorological data must be collected using a meteorological monitoring program compliant with RG 1.23. The data is then processed into the required input format for the atmospheric dispersion analysis code, ARCON 2.0. As discussed in Section 4.4.2 of NUREG/CR-6331, the inputs required by ARCON 2.0 include:

1) Lower instrumentation wind speed
2) Higher instrumentation wind speed
3) Lower instrumentation wind direction
4) Higher instrumentation wind direction
5) Wind stability

The wind stability is determined based on temperature differences captured by the lower and higher instruments. Temperature difference per unit length is then converted to stability class based on Table 4.4 (reproduced from Table 1 of RG 1.23).

Table 4.4 Classification of Atmospheric Stability

Stability Classification	Pasquill Stability Category	Ambient Temperature Change With Height ($°\underline{C/100m}$)
Extremely unstable	A	$\Delta T \leq -1.9$
Moderately unstable	B	$-1.9 < \Delta T \leq -1.7$
Slightly unstable	C	$-1.7 < \Delta T \leq -1.5$
Neutral	D	$-1.5 < \Delta T \leq -0.5$
Slightly stable	E	$-0.5 < \Delta T \leq 1.5$
Moderately stable	F	$1.5 < \Delta T \leq 4.0$
Extremely Stable	G	$\Delta T > 4.0$

4.4.6.3 Statistical Confidence Level

For evaluation of the EAB and LPZ receptor locations, consistent with RG 1.249, the more conservative of the 95th percentile overall or 99.5th percentile sector-wise χ/Q is used to ensure the assessment is bounding. The 95th percentile overall χ/Q is calculated using the distance from the nuclear island to the nearest point on the site boundary. For a multi-unit site, the distances from any nuclear island to the nearest point on the site boundary are considered; the minimum distance is used. For the 99.5th percentile, source-to-receptor distances are determined as described in subsection 4.4.6.4.

For evaluation of the MCR and TSC receptor locations, RG 1.194 is followed by selecting the 95th percentile χ/Q. Source-to-receptor distances and directions are determined as described in subsection 4.4.6.5.

4.4.6.4 Source-to-Receptor Distances and Directions for EAB and LPZ

ARCON 2.0 is applicable for atmospheric dispersion analysis when source-to-receptor distance falls between 10 m and 1200 m. The minimum distance limit is indicated in RG 1.194 and the maximum distance limit is indicated in RG 1.249.

The source-to-receptor distance and direction calculation methodology for the EAB and LPZ outer boundary follows the guidance in Regulatory Position 2.2.1 of RG 1.249. For the SMR-300 plant design, the EAB and LPZ outer boundary will coincide with site boundary. The distance calculation will follow one of the two approaches outlined in RG 1.249 and repeated below:

- First approach: In this approach, the limiting distance from the nuclear island to the nearest point on the site boundary is used for atmospheric dispersion analysis. For a multi-unit site, the distances from any nuclear island to the nearest point on the site boundary are considered; the minimum distance is used. Sixteen sets of χ/Q are generated per averaging time period – one for each of the 16 wind directions – at that single distance. From these sets of χ/Q, the largest value for each averaging period is selected. This results in a single set of bounding χ/Q (one per averaging period) to be used as input to RADTRAD for the radionuclide transport analysis. Figure 4-3 illustrates an example for the first approach.

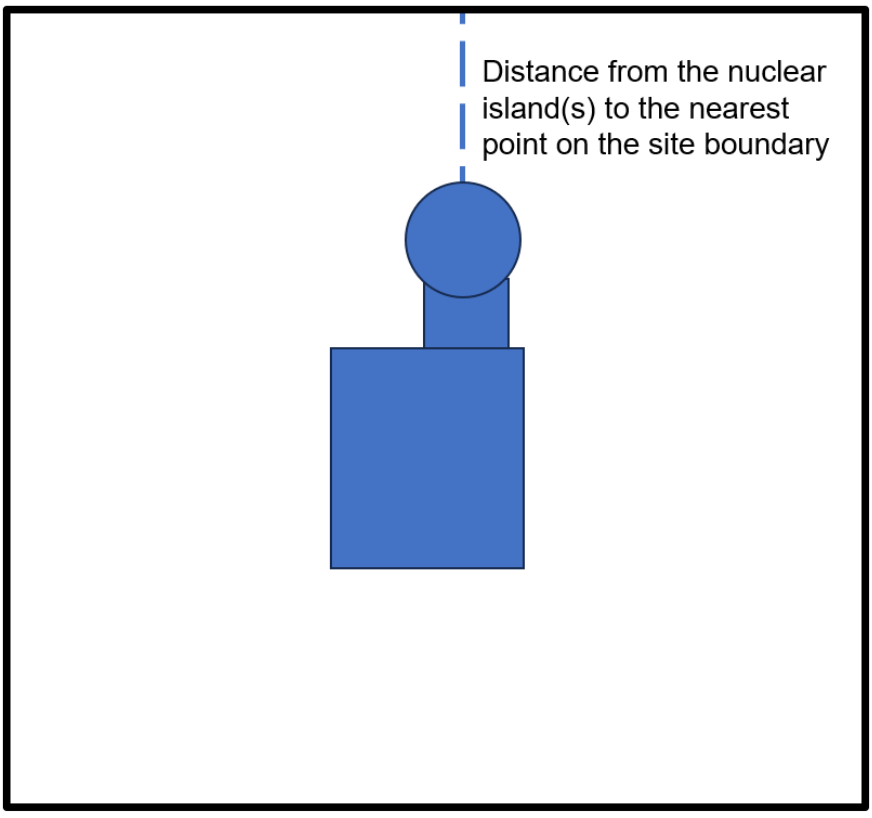


Figure 4-3 Example of Calculating Distance from the Nuclear Island to the Nearest Point on the Site Boundary

- Second approach: In this approach, the limiting distance between the nuclear island and the nearest point on the site boundary is evaluated in each of the 16 directional sectors (22.5° each). For a multi-unit site, the distances from any nuclear island to the nearest point on the site boundary for each sector are considered. The minimum source-to-receptor distance for each sector is determined within a 45° window centered on the direction of interest. This approach generates 16 sets of χ/Q per averaging period, from which the largest values for each period are identified. Figure 4-4 illustrates an example for the second approach.

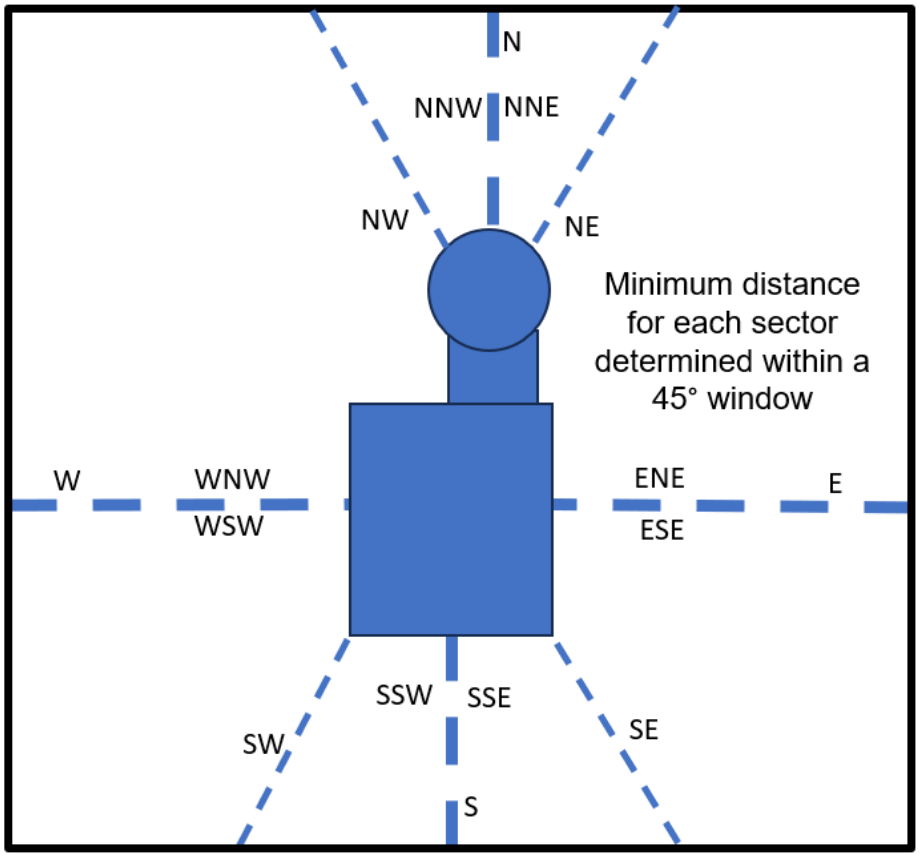


Figure 4-4 Example of Calculating Minimum Distance for Each Sector Within a 45° Window Centered on the Direction of Interest

4.4.6.5 Source-to-Receptor Distances and Directions for MCR and TSC

The source-to-receptor distance and direction calculation methodology for the MCR and TSC follows the guidance in Regulatory Position 3.4 of RG 1.194. First, all release points and air intakes are identified. The receptor is considered to be the point of air intake. The source-to-receptor distance is the shortest horizontal distance between the release point and the intake. The direction is the wind direction that would carry the plume from the release point to the intake. For each combination of release points and intakes, the minimum source-to-receptor distance and the direction are determined. The largest χ/Q for each averaging time period is then identified among all release point and intake combinations.

4.4.6.6 Release Characteristics

The release characteristics modeled in ARCON 2.0 for the EAB, LPZ, MCR, and TSC are described below:

- A ground level release mode is used. This is conservative as no credit is taken for elevated release. This is consistent with Regulatory Position 2.3.1 of RG 1.249 and Regulatory Position 3.2.1 of RG 1.194.
- A point source method is used to produce conservative estimates, which is consistent with Regulatory Position 2.3.4 of RG 1.249.
- A wind direction window of 90 degrees is used for MCR and TSC, which is consistent with Table A-2 of RG 1.194.
- A wind direction window of 360 degrees is used for the 95th percentile overall χ/Q calculation for EAB and LPZ, which is consistent with Regulatory Position 2.4.1 of RG 1.249.
- A wind direction window of 45 degrees is used for the 99.5th percentile sector-wise χ/Q calculation for EAB and LPZ, which is consistent with Regulatory Position 2.4.2 of RG 1.249.
- An elevation difference (difference between release height and intake height) of 0 m is used. Assuming the height at the same elevation is conservative, as ARCON 2.0 calculates the distance as the shortest line-of-sight distance, as mentioned in Footnote 1 to Regulatory Position 2.2.1 of RG 1.249.
- A building area of 0.01 m^2 is used. This is the smallest input that ARCON 2.0 accepts. Selecting the smallest building area is a conservative assumption as no credit is taken for enhanced diffusion from building wake.
- A surface roughness length of 0.2 m is used, which is consistent with Table A-2 of RG 1.194.
- A minimum wind speed of 0.5 m/s is used , which is consistent with Table A-2 of RG 1.194.
- An averaging sector width constant of 4.3 is used, which is consistent with Table A-2 of RG 1.194.
- Since no credit is taken for initial diffusion, the values for initial diffusion coefficients (σ_y,σ_z) are set to 0 m, which is consistent with Table A-2 of RG 1.194.
- Hours in averaging periods are set to the code default values, which is consistent with Table A-2 of RG 1.194.
- Minimum number of hours in periods are set to the code default values (up to 10% missing data), which is consistent with Table A-2 of RG 1.194.

4.4.6.7 Ground Deposition During Atmospheric Transport

Consistent with Regulatory Position 4.1 of RG 1.183, depletion of radioactive materials in the plume due to ground deposition is not credited.

4.4.7 MCR Habitability

For the MHA LOCA analysis, radionuclide transport through the systems that support MCR habitability is credited. The key design features of the SMR-300 systems supporting MCR habitability are summarized as follows:

- The non-safety-related non-radiologically controlled area heating, ventilation, and air conditioning (NRV) system automatically isolates normal air intake and routes air through a filtration unit when radiation levels in the outside air intake exceed a predefined threshold, as detected by a radiation monitor.
- The non-safety-related breathing air and pressurization (BAP) system provides an emergency source of pressurized air and means for recirculation through a filter unit to maintain a habitable environment for 72 hours when the NRV is unavailable. This ensures that the control room remains habitable when the NRV is unavailable, including during a loss of offsite power.
- After the 72-hour emergency operation period, it is assumed that the NRV can be reactivated to maintain habitable control room conditions.
- The ventilation systems and MCR are designed to minimize in-leakage and maintain the Control Room Emergency Zone at a positive pressure relative to surrounding areas.
- The exhaust flow rate is assumed to equal the total inlet airflow in both normal and emergency modes to maintain pressure balance, ensuring that the same volume of air is expelled as is taken in.

4.4.8 TSC Habitability

For the MHA LOCA analysis, radionuclide transport through the systems that support TSC habitability is credited. The key design features of the SMR-300 systems supporting TSC habitability are summarized as follows:

- The non-safety-related normal TSC ventilation system automatically isolates normal air intake, routes air through a filtration unit and activates recirculation when radiation levels in the outside air intake exceed a predefined threshold, as detected by a radiation monitor.
- The TSC ventilation system is designed to minimize in-leakage and maintain the TSC at a positive pressure relative to surrounding areas.
- The exhaust flow rate is assumed to equal the total inlet airflow in both normal and emergency modes to maintain pressure balance, ensuring that the same volume of air is expelled as is taken in.

4.5 Dose Consequences

Dose can be accumulated through radionuclide transport (immersion, inhalation), and radiation transport (other shine contributions).

RADTRAD serves as the computational framework to evaluate TEDE due to radionuclide transport. It takes inputs for radionuclide inventory, atmospheric dispersion, natural aerosol deposition, elemental wall deposition, SFP scrubbing, containment leakage, partitioning decontamination, and other factors. The TEDE calculated by RADTRAD combines contributions from immersion dose and inhalation dose. Immersion doses arise from external exposure to a radioactive plume enveloping the individual, while inhalation doses result from radioactive particles or gases being inhaled and deposited internally.

Consistent with RG 1.183, EAB and LPZ dose evaluations consider immersion and inhalation doses only. Other possible radiation shine contributions (e.g., from radioactivity within containment) are not considered; the dose contribution is negligible because of the distance between the sources and receptors.

Consistent with RG 1.183, MCR and TSC dose evaluations consider immersion, inhalation, and other radiation shine doses, as described in subsection 4.5.2.2.

4.5.1 Dose Consequences for EAB and LPZ

Consistent with Regulatory Position 4.1.e of RG 1.183, dose calculations using RADTRAD determine the TEDE received by the most limiting receptor at the most limiting EAB location. The maximum 2-hour TEDE is determined by calculating the postulated dose for a series of time increments and performing a “sliding” sum over the increments for successive 2-hour periods.

Consistent with Regulatory Position 4.1.f of RG 1.183, dose calculations using RADTRAD determine the TEDE received by the most limiting receptor at the outer boundary of the LPZ for the duration of the accident.

4.5.1.1 Breathing Rates

Consistent with Regulatory Position 4.1 of RG 1.183, the breathing rates for the most limiting receptors at the EAB and LPZ are provided in Table 4.5.

Table 4.5 Breathing Rates for EAB and LPZ

Location	Duration (hours)	Breathing Rates (m^3/s)
EAB	0-720	3.5×10^{-4}
LPZ	0-8	3.5×10^{-4}
	8-24	1.8×10^{-4}
	24-720	2.3×10^{-4}

4.5.1.2 Dose Calculation

The dose to a hypothetical individual at the EAB or LPZ is calculated using the specified χ/Qs and the amount of each nuclide released during the exposure period.

In RADTRAD, the air immersion dose from each nuclide, n, in an environment compartment is calculated as:

$$\boldsymbol{D_{c,n}^{env} = A_n \cdot (\chi/Q) \cdot DCF_{c,n}} \qquad \textbf{Equation 4-7}$$

where,

$D_{c,n}^{env}$ = Air immersion (cloud-shine) dose due to nuclide n in the environment compartment $[Sv]$

A_n = Released activity of nuclide n $[Bq]$

χ/Q = Atmospheric dispersion factors $[\frac{s}{m^3}]$

$DCF_{c,n}$= Air immersion (cloud-shine) DCF for nuclide n $[\frac{Sv \cdot m^3}{Bq \cdot s}]$ (obtained from Table III.1 of FGR 12 [21])

In RADTRAD, the inhalation dose from each nuclide, n, is calculated as:

$$\boldsymbol{D_{i,n}^{env} = A_n \cdot (\chi/Q) \cdot BR \cdot DCF_{i,n}} \qquad \textbf{Equation 4-8}$$

where,

$D_{i,n}^{env}$ = Inhalation dose commitment due to nuclide n in the environment compartment $[Sv]$

BR = Breathing rate $[\frac{m^3}{s}]$

$DCF_{i,n}$= Inhalation DCF for nuclide n $[\frac{Sv}{Bq}]$ (obtained from Table 2.1 of FGR 11)

4.5.2 Dose Consequences for MCR and TSC

Consistent with Regulatory Position 4.2.1 of RG 1.183, the radiological consequences methodology considers all sources of radiation that contribute to TEDE at receptors in the MCR and the TSC. The following sources of radiation are considered:

1) Contamination of the MCR or TSC atmosphere by the intake or infiltration of the radioactive material contained in the radioactive plume released from the facility.
2) Contamination of the MCR or TSC atmosphere by the intake or infiltration of airborne radioactive material from adjacent areas and structures.
3) Radiation shine from the external radioactive plume released from the facility.
4) Radiation shine from radioactive material in the reactor containment.
5) Radiation shine from radioactive material in systems and components internal or external to the MCR and TSC (e.g., radioactive material buildup in recirculation filters).

Sources 1 and 2 are addressed through radionuclide transport analysis by accounting for inhalation and immersion doses. Radiation transport analysis is used to evaluate shine dose contributions from sources 3, 4, and 5.

The TEDE is determined for the most limiting receptors in the MCR and TSC for the duration of the accident. Dose mitigation through the NRV or BAP systems is not credited for any DBAs

except the MHA LOCA. For MHA LOCA, either the NRV or BAP is assumed to be in operation and their actuation time is accounted for conservatively.

4.5.2.1 Immersion and Inhalation Dose

4.5.2.1.1 Breathing Rates

Consistent with Regulatory Position 4.2.6 of RG 1.183, the breathing rates for the most limiting receptors in the MCR and TSC, are provided in Table 4.6.

Table 4.6 Breathing Rates for MCR and TSC

Location	Duration (hours)	Breathing Rates (m^3/s)
MCR	0-720	3.5×10^{-4}
TSC	0-720	3.5×10^{-4}

4.5.2.1.2 Occupancy Factors

Consistent with Regulatory Position 4.2.6 of RG 1.183, the dose receptor for the MCR and TSC analyses is the hypothetical maximum exposed individual who is present in the location for a duration that considers the occupancy factors in Table 4.7. The duration of exposure is calculated by multiplying the number of hours from initiation of the event by the corresponding occupancy factors and then summing the results for the event duration.

Table 4.7 Occupancy Factors

Time (hours)	Occupancy factor
0-24	1.0
24-96	0.6
96-720	0.4

4.5.2.1.3 Dose Mitigation Equipment and Medicines

Personal protective equipment, including beta radiation-resistant clothing, eye protection, and self-contained breathing apparatus, are not credited for the dose analysis. Prophylactic measures such as potassium iodide pills are also not credited.

4.5.2.1.4 Direct Release Path

There are multiple barriers between the radiologically controlled area of the Reactor Auxiliary Building (RAB) and the non-radiologically controlled area of the RAB where the MCR is located, including fire-rated metal doors, along with administrative controls regulating entry and exit through these doors. The Turbine Building (TB) is physically separated from the RAB, with no interconnecting passageways. Therefore, no direct pathway is considered for radiological release from the radiologically controlled area of the RAB or TB to the MCR.

The building housing the TSC is an independent structure, separate from both the RAB and the TB. Therefore, no direct pathway is considered for radiological release from the radiologically controlled area of the RAB or TB to the TSC.

4.5.2.1.5 Immersion and Inhalation Dose Calculation

The dose to a hypothetical individual in the MCR or TSC is calculated based on the time-integrated concentration of radionuclides in the compartment.

In RADTRAD, the air immersion dose from each nuclide, n, in the MCR or TSC is calculated as:

$$\boldsymbol{D_{c,n}^{CR} = \int C_n(t) \cdot dt \cdot \left(\frac{DCF_{c,n}}{G_F}\right) \cdot OF} \qquad \textbf{Equation 4-9}$$

where,

$D_{c,n}^{CR}$ = Air immersion (cloud-shine) dose due to nuclide n in the control room (MCR or TSC) compartment $[Sv]$

$C_n(t)$ = Instantaneous concentration of radionuclide n in the compartment $[Bq(s)]$

G_F = Murphy-Campe geometric factor [22]

$= 1173/V^{0.338}$

$V =$ Volume of the MCR or TSC $[ft^3]$

OF = Occupancy factor

In RADTRAD, the inhalation dose from each nuclide, n, in the MCR or TSC is calculated as:

$$\boldsymbol{D_{i,n}^{CR} = \int C_n(t) dt \left(BR \cdot OF \cdot DCF_{i,n}\right)} \qquad \textbf{Equation 4-10}$$

where,

$D_{i,n}^{CR}$ = Inhalation dose commitment due to nuclide n in the MCR or TSC compartment $[Sv]$

4.5.2.2 Radiation Transport

To evaluate shine dose contributions, RADTRAD is used to conservatively calculate radioactive material concentration in various compartments. These concentrations serve as input to ORIGAMI (a SCALE module) and MCNP models for further analysis to determine radiation shine dose.

A bounding analysis approach is adopted rather than performing individual calculations for each accident scenario. The bounding accident scenario is selected based on the scenario with the largest activity in the containment, MCR, and TSC. The resulting bounding shine dose is combined with inhalation and immersion doses for MCR and TSC personnel during the accident to determine the TEDE.

5.0 CONCLUSION

This topical report describes and substantiates a methodology that can be employed to analyze the radiological consequences of DBAs for the SMR-300 design. The site-independent methodology incorporates the use of computational codes and justified assumptions to assess the TEDE for receptors at the EAB, at the LPZ outer boundary, in the MCR, and in the TSC. The methodology conforms to NRC regulatory guidance provided in RG 1.194 and RG 1.249 and largely conforms to the guidance in RG 1.183. The deviations from the guidance in RG 1.183 are summarized as follows:

- Application of Assumption A-1.1 of RG 1.183 for iodine chemical forms in MHA LOCA analyses when containment flood-up pool pH ≥ 6.
- Application of Assumption H-4 of RG 1.183 for iodine chemical forms in REA analyses when containment flood-up pool pH ≥ 6.
- Application of Assumption B-2 of RG 1.183 for elemental iodine DF computation for Phase 1 release in FHA analyses when SFP water depth is greater than 23 feet.

Additional aspects of the methodology that adhere to NRC regulatory guidance, but could be considered unique, include:

- Determination of the “at-instant” core radionuclide inventory, consistent with Regulatory Position 3.1 of RG 1.183. The “at-instant” inventory represents radionuclide activity values at the instant of an accident, which is chosen to be bounding for the analyzed accident type.
- Application of the model in NUREG/CR-6189 to determine the 10th percentile decontamination coefficient for natural aerosol deposition inside containment.
- Application of the equation in NUREG-0800, Section 6.5.2 for the elemental iodine removal coefficient for natural wall deposition inside containment.
- Application of iodine spiking assumptions from RG 1.183 for secondary plant release path accidents with no fuel breach due to either cladding failure or fuel melt.

The methodology can be used to demonstrate that the SMR-300 design is compliant with the NRC regulatory requirements in § 50.34(a)(1)(ii)(D), § 50.34(b)(11), and GDC 19 in Appendix A to 10 CFR Part 50.

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