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SMR-300
Accident Radiological Consequences Methodology

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Meeting Agenda

- Purpose of Meeting
- Topical Report Scope & Applicable Regulatory Guidance
- SMR-300 Design Overview

Purpose

- To provide an overview of the methodology for assessing radiological consequences of SMR-300 design basis accidents. The methodology will be used to:
  - Calculate radiation doses for EAB and LPZ boundary determination
  - Calculate radiation doses to the MCR and TSC
  - Meet the intent of 10 CFR 50.34(a)(1)(ii)(D) and 10 CFR 50.34(b)(11)
- The methodology is generalized and non-site-specific

Topical Report Scope

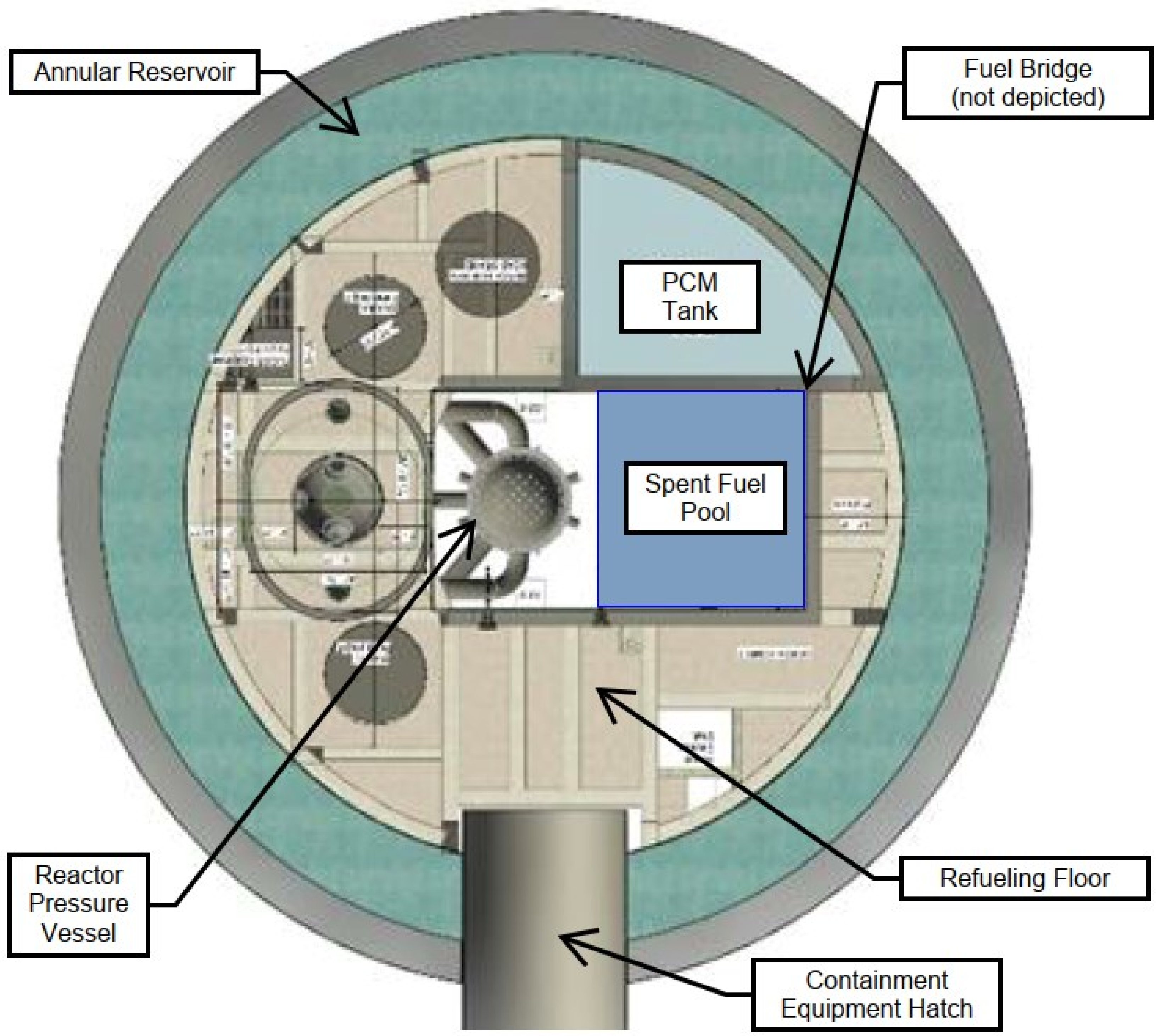
- Seeking approval for overall methodology, which largely follows available Regulatory Guides, except for:
  - Use of **RG 1.183 R1 Assumption A-1.1** iodine chemical forms for pH > 6
  - Credit of partial flashing in steaming region of OTSG, inconsistent with **RG 1.183 R1 Assumption E-6.5**.
  - Use of sector-specific 95th percentile atmospheric dispersion coefficients for EAB and LPZ (based on **RG 1.194 R0**, inconsistent with **RG 1.249 R0**)
- Outside the scope of this topical report:
  - EPZ sizing methodology
  - Methodology to determine failed fuel fractions
  - Methodology for accident pH analysis

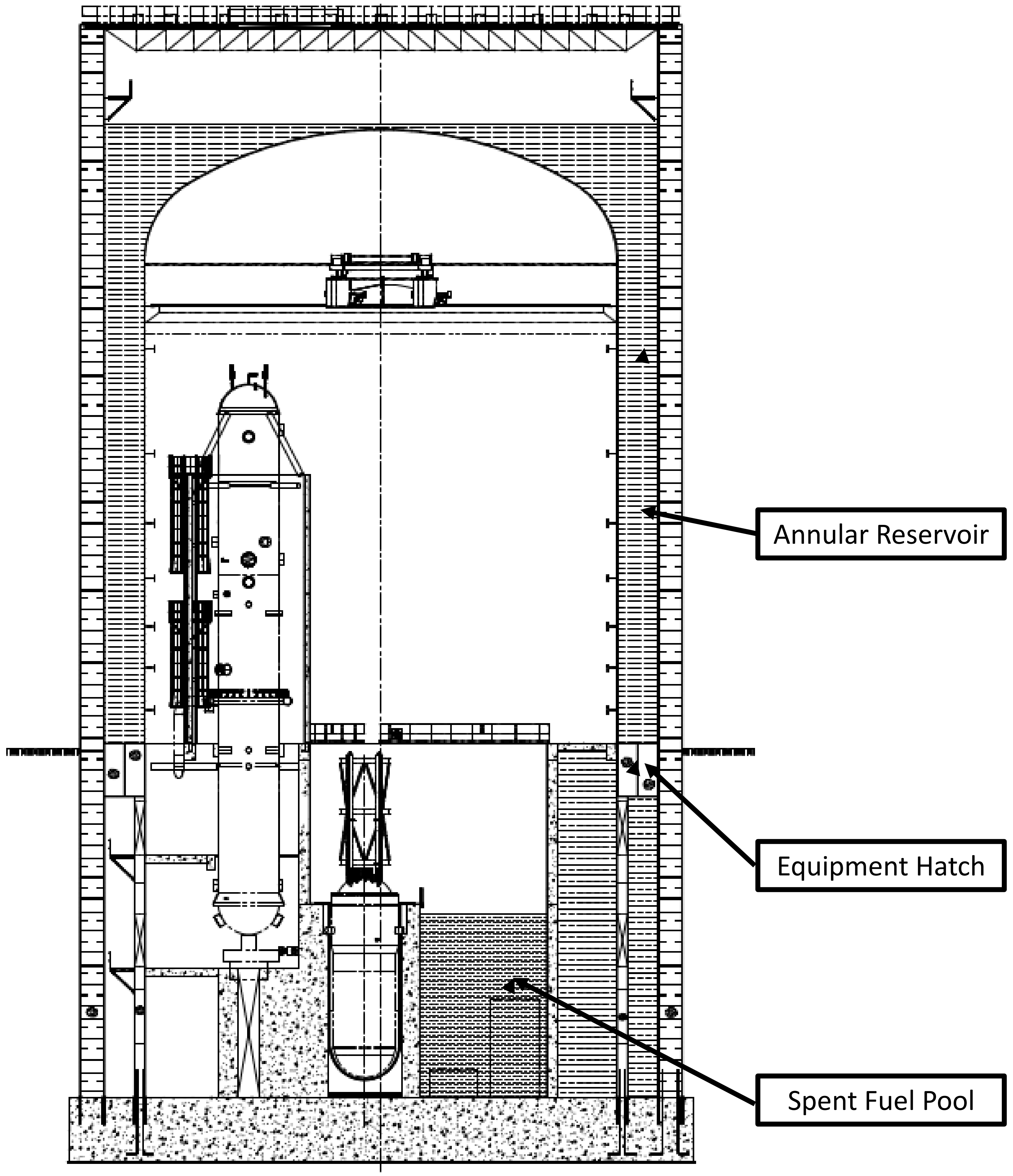
Applicable Regulatory Guidance

- Regulatory Guides 1.183 R1, 1.194 R0, and 1.249 R0
  - Following unless otherwise noted and justified
- NUREG/CR-5950: Iodine Evolution and pH Control
  - Basis for iodine evolution for containment pH ≥ 6
- NUREG-0933 Issue 197: Iodine Spiking Phenomena
  - Basis for iodine release rate used to calculate design basis reactor coolant inventory
- NUREG/CR-6189: A Simplified Model of Aerosol Removal by Natural Processes in Reactor Containments
  - Acceptable natural aerosol deposition model (Powers Deposition Model); integrated into RADTRAD
- NUREG-0800, Section 6.2.6 R3: Containment Leakage System
  - Basis for minimum acceptable design containment leakage rate
- NUREG-0800, Section 6.5.2 R3, Containment Spray as a Fission Product Cleanup System
  - Basis for elemental iodine removal by natural wall deposition (SMR-300 does not have containment spray)

SMR-300 Design Overview

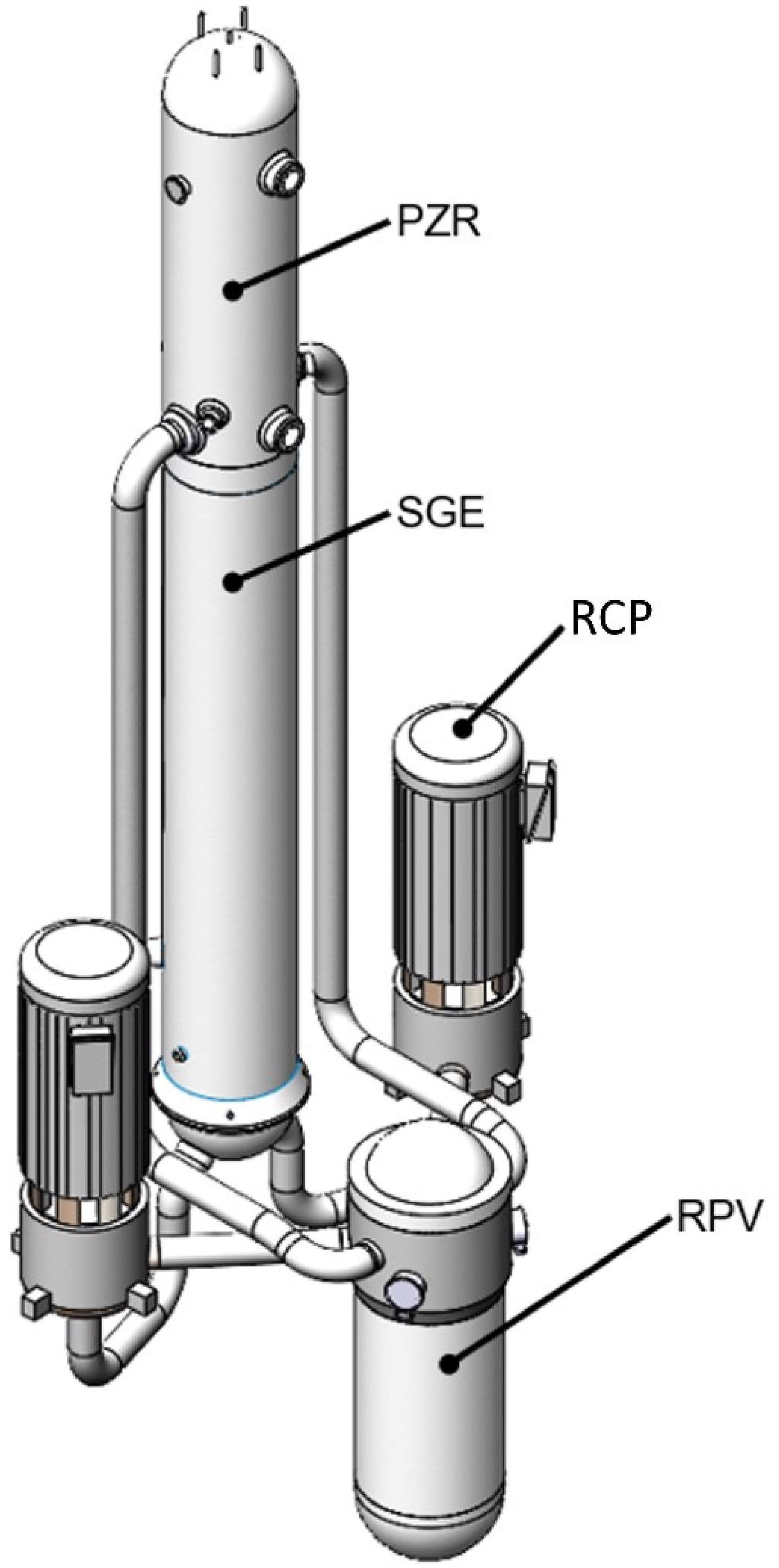
- Relevant Unique Features:
  - ✓ Spent Fuel Pool inside containment
  - ✓ Annular Reservoir surrounding above grade containment

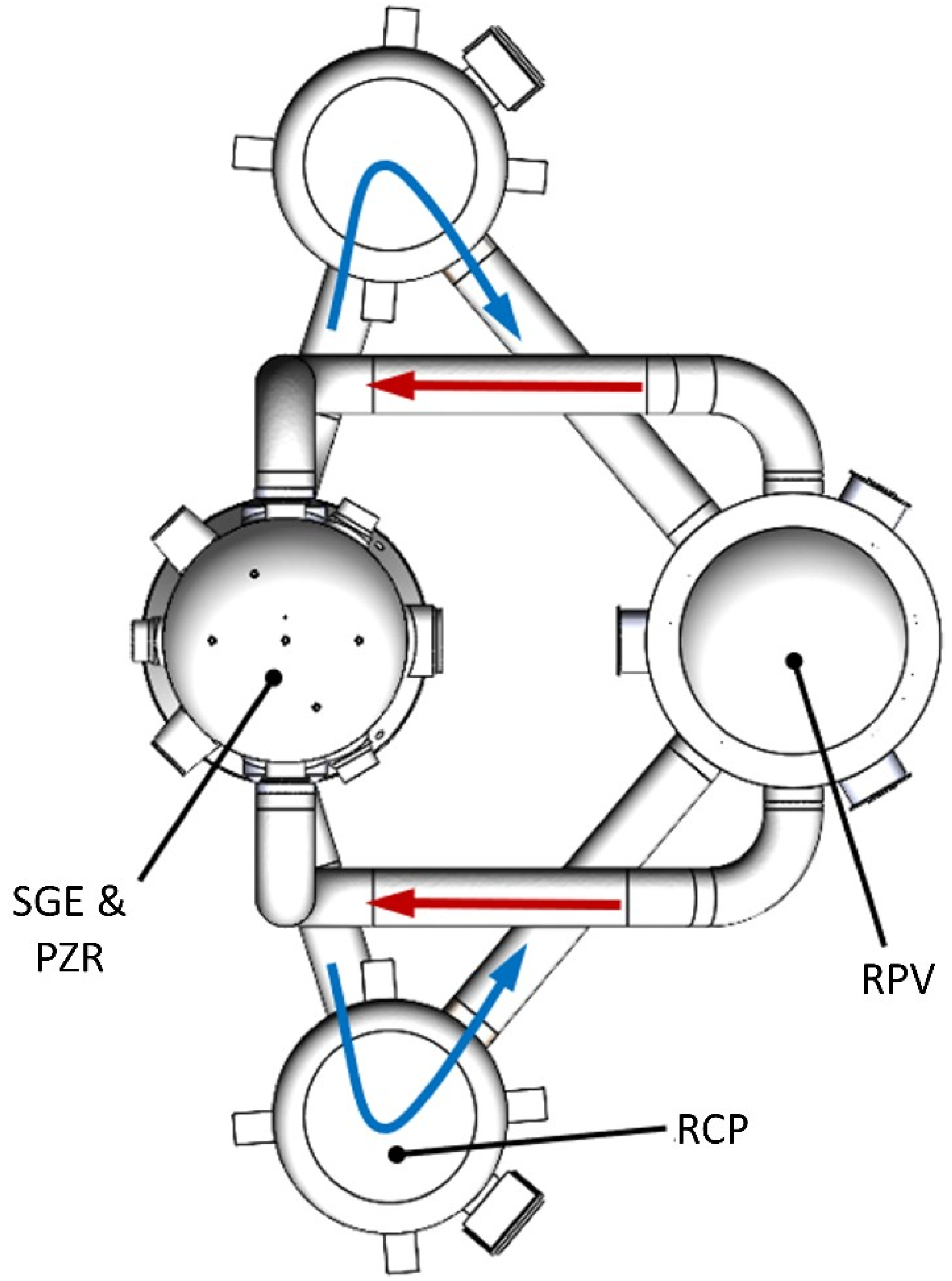




SMR-300 Design Overview

- Relevant Unique Features:
  - Once-Through Steam Generator





Methodology

Accident Duration

Receptor Location	Time Duration [1]	References
Exclusion Area Boundary	Worst 2 hours in 720 hours (30 days)	Table 7, Regulatory position 4.1.e, RG 1.183 R1
Low Population Zone	720 hours (30 days)	Table 7, Regulatory Position 4.1.f, RG 1.183 R1
Main Control Room	720 hours (30 days)	Table 7 note 2 of RG 1.183 R1
Technical Support Center	720 hours (30 days)	Table 7 note 2 of RG 1.183 R1

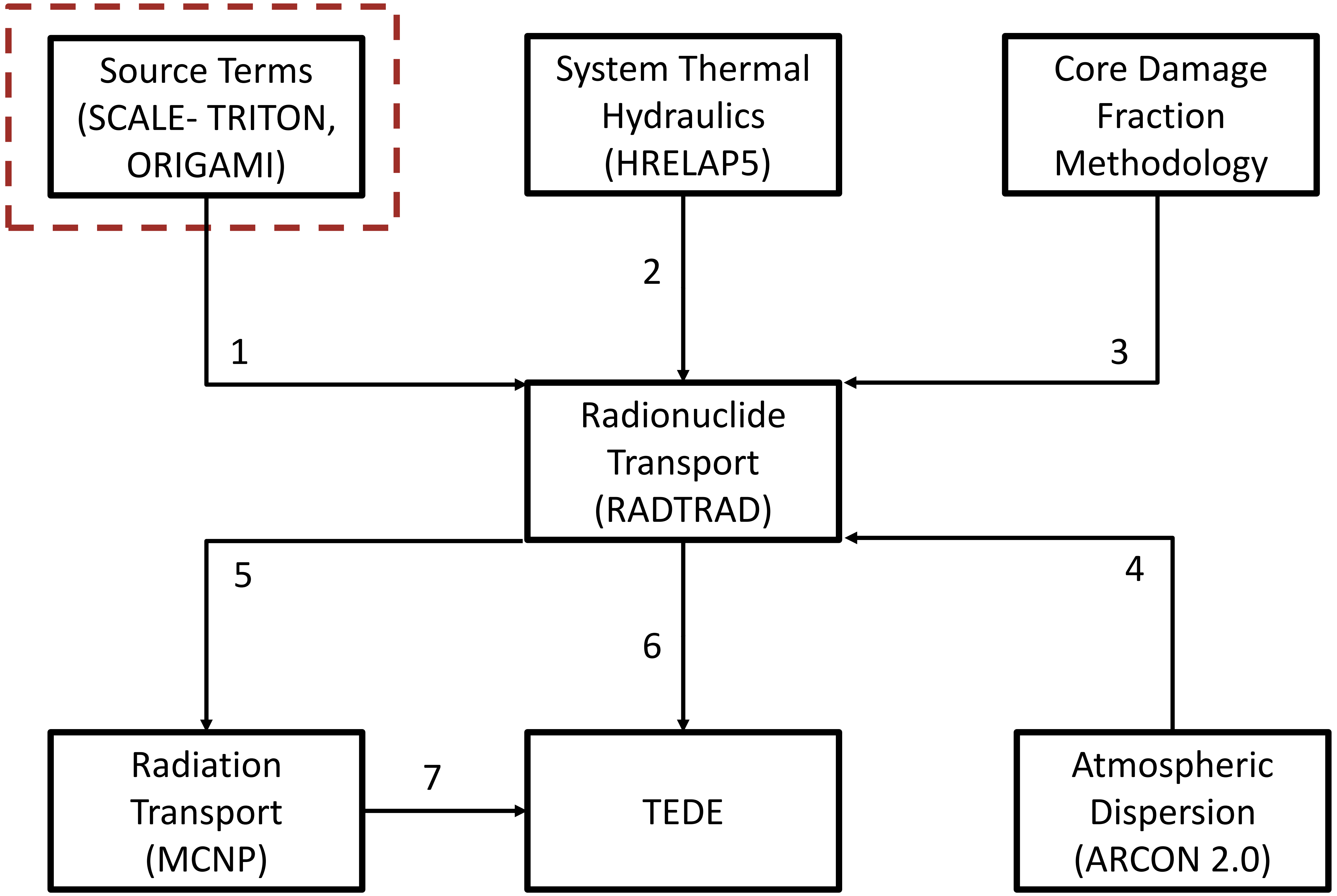
Notes:

1. SGTR, REA, REA-Secondary path release accident duration will be conservatively bounded by 30 days.

Dose Acceptance Criteria

Accident	EAB and LPZ Dose Criteria (TEDE)	MCR and TSC Dose Criteria (TEDE)
MHA LOCA	25 rem	5 rem
DBA LOCA	25 rem	5 rem
SGTR		
Pre-Accident Spike	25 rem	5 rem
Concurrent Spike	2.5 rem	5 rem
MSLB		
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	6.3 rem	5 rem

Analysis Flowchart: Inventory Calculations



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, Plum and Filters Inventory
6	Inhalation and Ingestion Dose
7	Radiation Shine Dose

Core Radionuclide Inventory

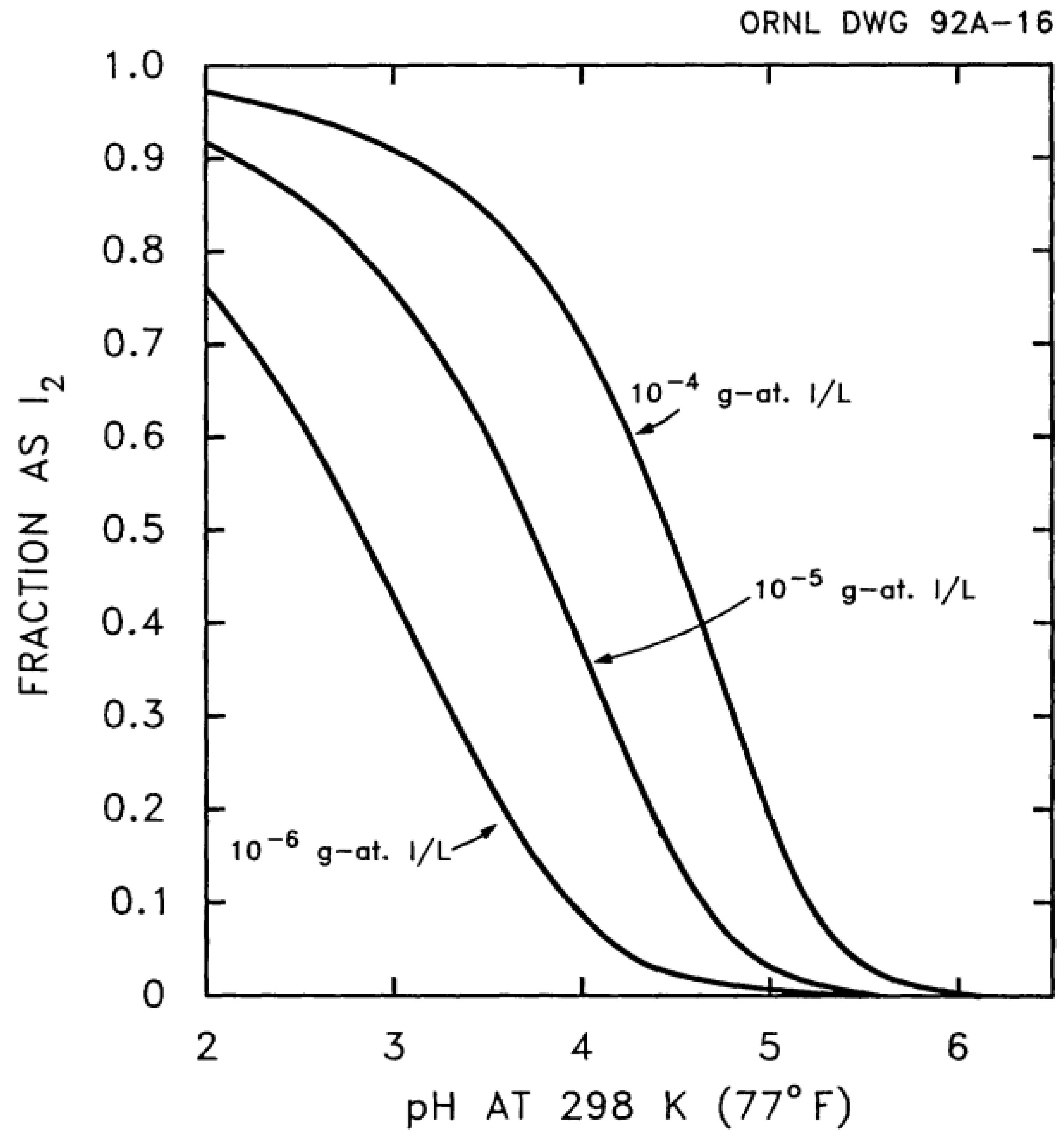
- **Regulatory Position 3.1 of RG 1.183 R1** followed for ‘at-instant’ core radionuclide inventory.
  - Activity values of radionuclides determined at the instant of the accident. The instant is chosen to be bounding for the accident type.
- Single assembly inventory is calculated based on:
  - Detailed geometry of the fuel assembly
  - Assembly Radial Power Factor (RPF)
  - Rated power with uncertainty
  - Assembly-average exposure
  - U-235 enrichment

Coolant Inventory

- Primary
  - Principal mechanism is fission product leakage into the coolant, conservatively determined based on failed fuel fraction
    - No core damage – highest fuel defect level allowed during normal operation
    - Core damage – based on cladding failure and/or core melt
- Secondary
  - Principal mechanism is primary leakage to secondary side

Source Terms: Accidents with Release from Containment

- LOCA
  - PWR release fractions from **Table 2** and timings from **Table 5 of RG 1.183 R1**
- REA Containment Release
  - All noble gases and half iodines and alkali metals assumed released from melted fuel in conformance with **Assumption H-1 of RG 1.183 R1**
- FHA
  - Only radionuclides in gap assumed released; gap release fractions from **Table 4 of RG 1.183 R1**
  - Instantaneous release timings conservatively assumed
- Chemical form fractions of iodine considered:
  - 95% cesium iodide
  - 4.85% elemental iodide
  - 0.15% organic iodide
- Although **Assumption A-1.1 of RG 1.183 R1** prescribes chemical forms for pH ≥ 7, SMR-300 is designed to maintain pH ≥ 6. **Figure 3.1 of NUREG/CR-5950** shows iodine evolution to elemental (I_2) form is much less than 4.85% for pH = 6; therefore, RG 1.183 R1 chemical form fractions are applicable to SMR-300 design.

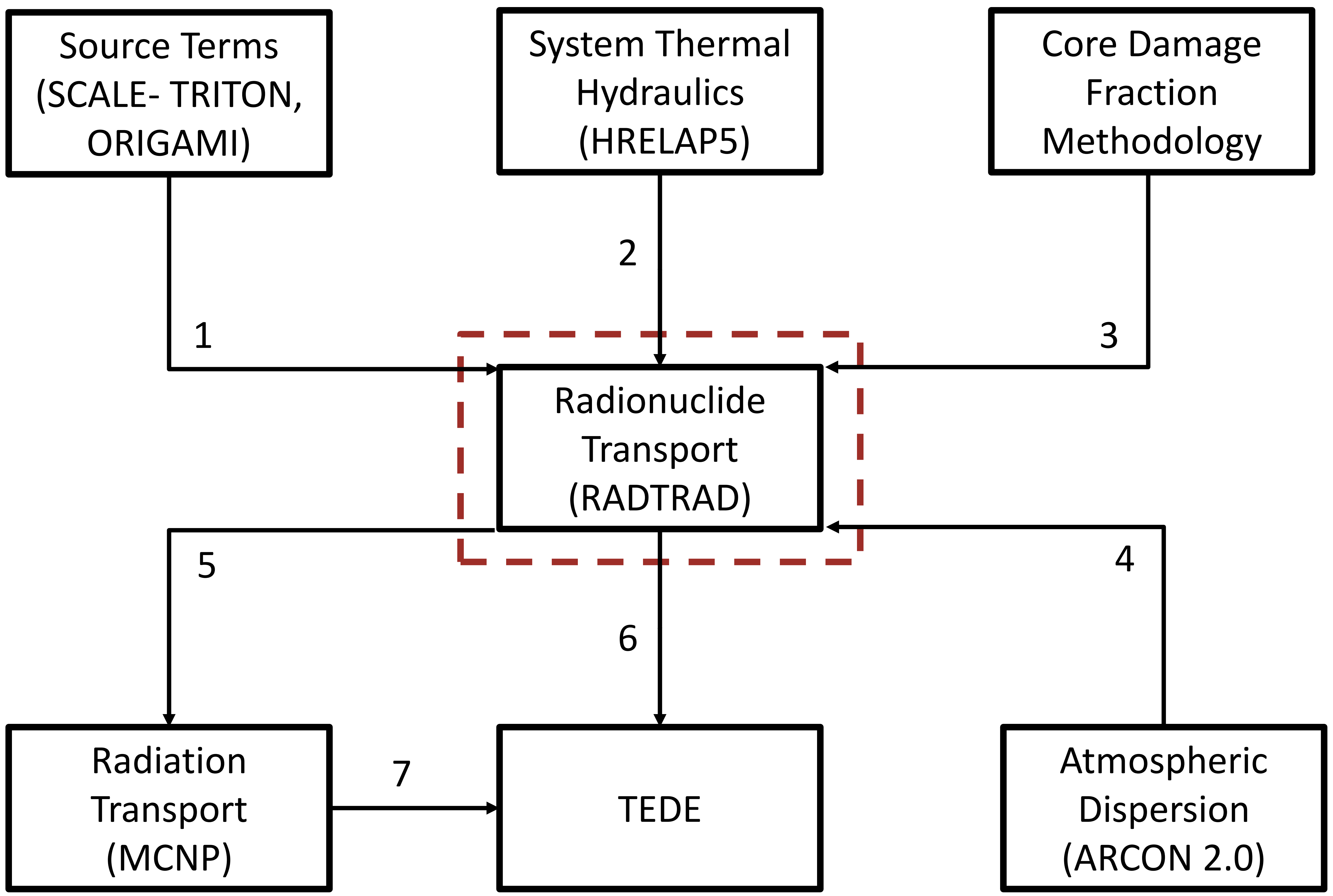


Model calculations of fraction as I_2 vs pH
Figure 3.1 of NUREG/CR 5950

Coolant Source Terms

- Accidents with No Fuel Breach
  - Only gap release considered given principal mechanism is quantity of fuel rods with defects; only noble gases, halogens and alkali metals isotopes considered. Release fractions from **Table 4 of RG 1.183 R1**.
  - Instantaneous release conservatively assumed
  - Chemical form fractions of iodines in conformance with **Assumptions E-5, F-5, and H-5 of RG 1.183 R1**
- Accidents with Fuel Breach (REA with Secondary Plant Release Path)
  - All noble gases and half iodines contained in the melted fraction are released to coolant in conformance with **Assumption H-1 of RG 1.183 R1**. This is in addition to gap release
  - Instantaneous release conservatively assumed

Analysis Flowchart: Radiological Transport



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Radionuclide Transport: Spent Fuel Decontamination

- Filtered pathway modeled to simulate retention of radioactive material in spent fuel water column following FHA in conformance with **RG 1.183 R1 Appendix B**
  - Retention of noble gases is negligible (decontamination factor = 1)
  - Full retention of particulate radionuclides assumed (decontamination factor is infinite)
  - Retention of elemental isotopes occurs in two phases:
    - Phase 1 (0-2 hours) – decontamination factor conservatively calculated using Assumption B-2 of RG 1.183 R1.
      - SMR-300 SFP depth > 23 ft
    - Phase 2 (2 hours – 30 days) – no decontamination credited

Radionuclide Transport: Natural Deposition

- Natural aerosol deposition credited for all accidents (including MHA LOCA) involving containment pathway releases in conformance with **Assumption A.2-2 of RG 1.183 R1**.
- Assumption A.2-2 of RG 1.183 R1 also allows credit to be taken for iodine removal due to iodine gas interaction with free surfaces inside containment. Removal coefficient determined using:

$$\lambda_d = \frac{K_W A}{V}$$

where,

V = containment building free volume, m^3

A = wetted surface area, m^2

K_w = 4.9 m/hr (NUREG-800, Section 6.5.2 R3)

Radionuclide Transport: Secondary Path Release

- Three sources of radioactive material release to environment considered:
  - Primary-to-secondary normal leakage (all accidents involving secondary path release)
    - Assumed as Limiting Condition of Operation (LCO) provided in SMR-300 Technical Specifications (TS)
    - Assumed release direct to environment without mitigation or retention
  - Primary-secondary break flow due to SGTR (SGTR accident only)
  - Steaming of steam generator bulk water (all accidents involving secondary path release)

Radionuclide Transport: Secondary Path Release (Cont.)

- HRELAP5 analysis used to determine:
  - SGTR break flow in steaming region
    - Only flashed portion assumed to leak to environment
    - Flashing fraction (deviates from intent of Assumption E.6-5 of RG 1.183 R1 given SMR-300 has OTSG):

$$\alpha = \frac{H_{RCS}^{L} - H_{SG}^{L}}{H_{SG}^{V} - H_{SG}^{L}}$$

  - Steam generator steaming rate
- Radioactivity in secondary water and non-flashed primary coolant is assumed to become vapor at a rate that is a function of steam rate and the partition coefficient (assumed 100)
- No credit taken for scrubbing in OTSG
- Credit applied for the decay of radionuclides, including the formation of decay daughters, until release to the environment

Radiological Transport: Control Room Habitability

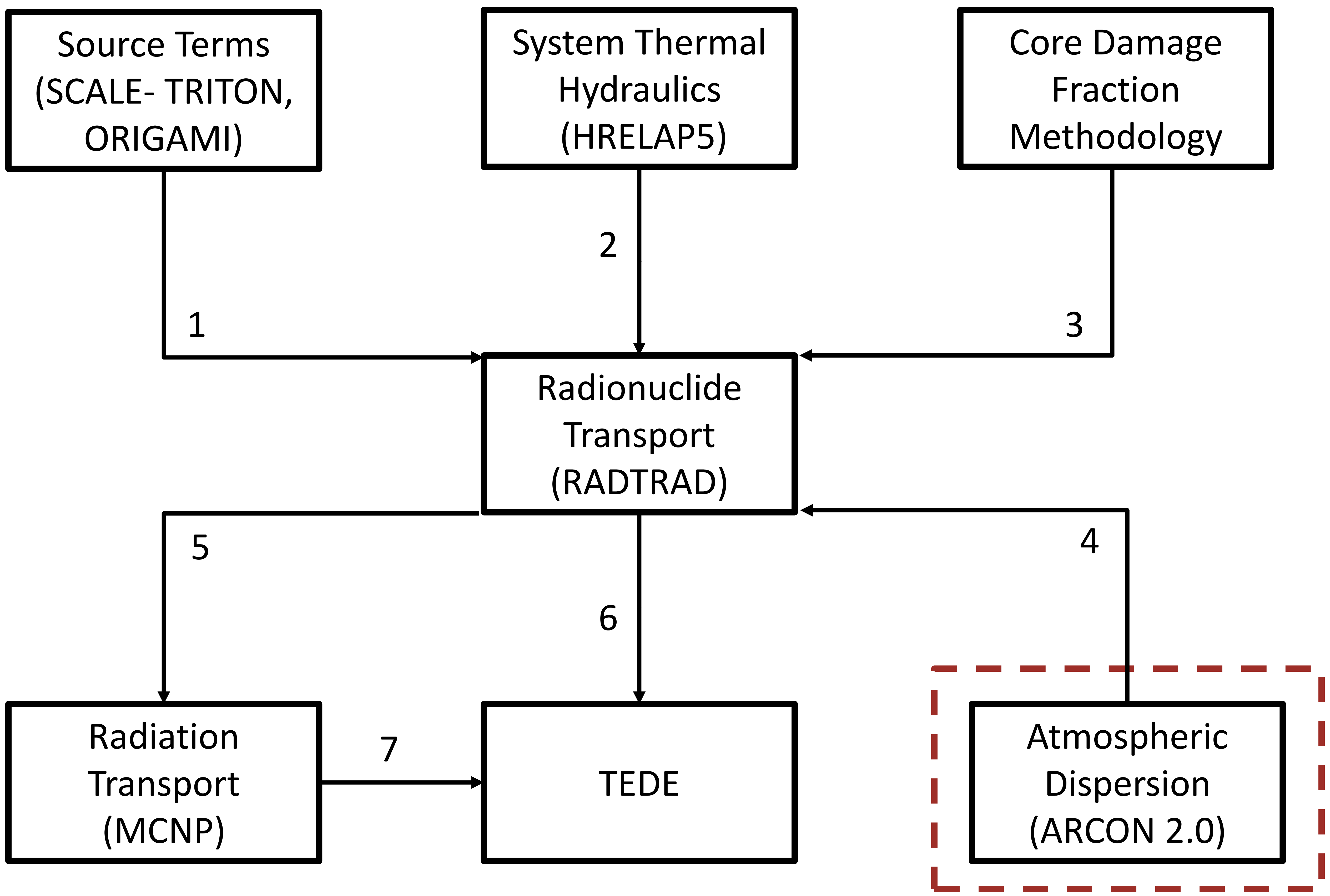
- Operation action not credited in SAR Chapter 15 (i.e., operators serve no safety-related function)
  - SMR-300 control room habitability is not a safety-related function
- SMR-300 control room habitability provided by two non-safety systems
  - Control Room Ventilation (CRV)
    - Normal HVAC to control room exclusion zone (CREZ); upon reaching high radiation set point, isolates normal air intake, routes air through filtration unit, and activates recirculation
  - Breathing Air and Pressurization (BAP) system
    - Isolates outside air and CREZ upon reaching very high radiation set point. CRV isolated, BAP provides emergency air for 72 hours
  - Both CRV and BAP maintain positive pressure in CREZ

Accident Specific Assumptions

- MHA LOCA[1]
  - Full core melt postulated as result of a LOCA
- FHA
  - Release to environment assumed through open airlock
- Locked Rotor/REA Containment Release
  - Extent of fuel damage due to cladding breach and fuel melt will be determined by methodology to determine failed fuel fractions
- SGTR/MSLB/REA Secondary Plant Release
  - During normal operations, primary coolant leaks/flashes in secondary side at allowable LCO rate
  - Activity in the primary coolant escaping to the secondary side is assumed to immediately be airborne due to flashing and atomization; released without mitigation

- [1] Separate, less conservative MHA assumptions are being considered for use in EPZ sizing methodology, which is outside the scope of this topical report

Analysis Flowchart: Atmospheric Dispersion



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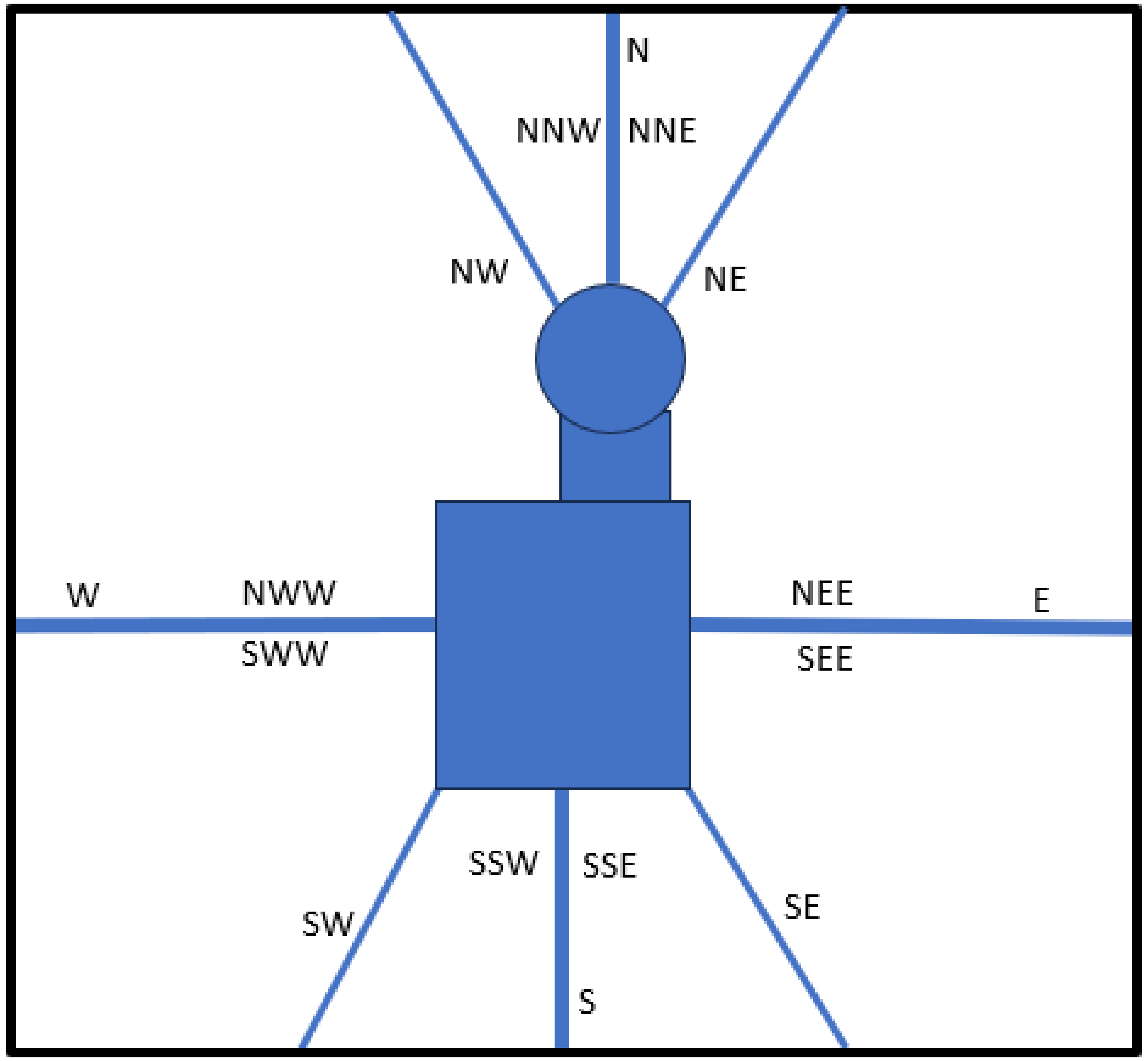
Atmospheric Dispersion

- No credit taken for depletion of radioactive materials in plume caused by decay or ground deposition
- **Regulatory Position 5.3 of RG 1.183 R1** identifies use of **RG 1.194 R0**, **RG 1.145 R1** and **RG 1.249 R0** for atmospheric dispersion factor calculations
  - SMR-300 radiological consequence methodology utilizes ARCON 2.0 to determine site-specific 95th percentile atmospheric dispersion coefficients for all dose calculations (consistent with **RG 1.194 R0**, inconsistent with **RG 1.249 R0**)
    - SMR-300 anticipates all potential future sites will have onsite EAB and LPZ
    - Quantification of plume dispersion will rely on site-specific meteorological data for atmospheric dispersion analysis

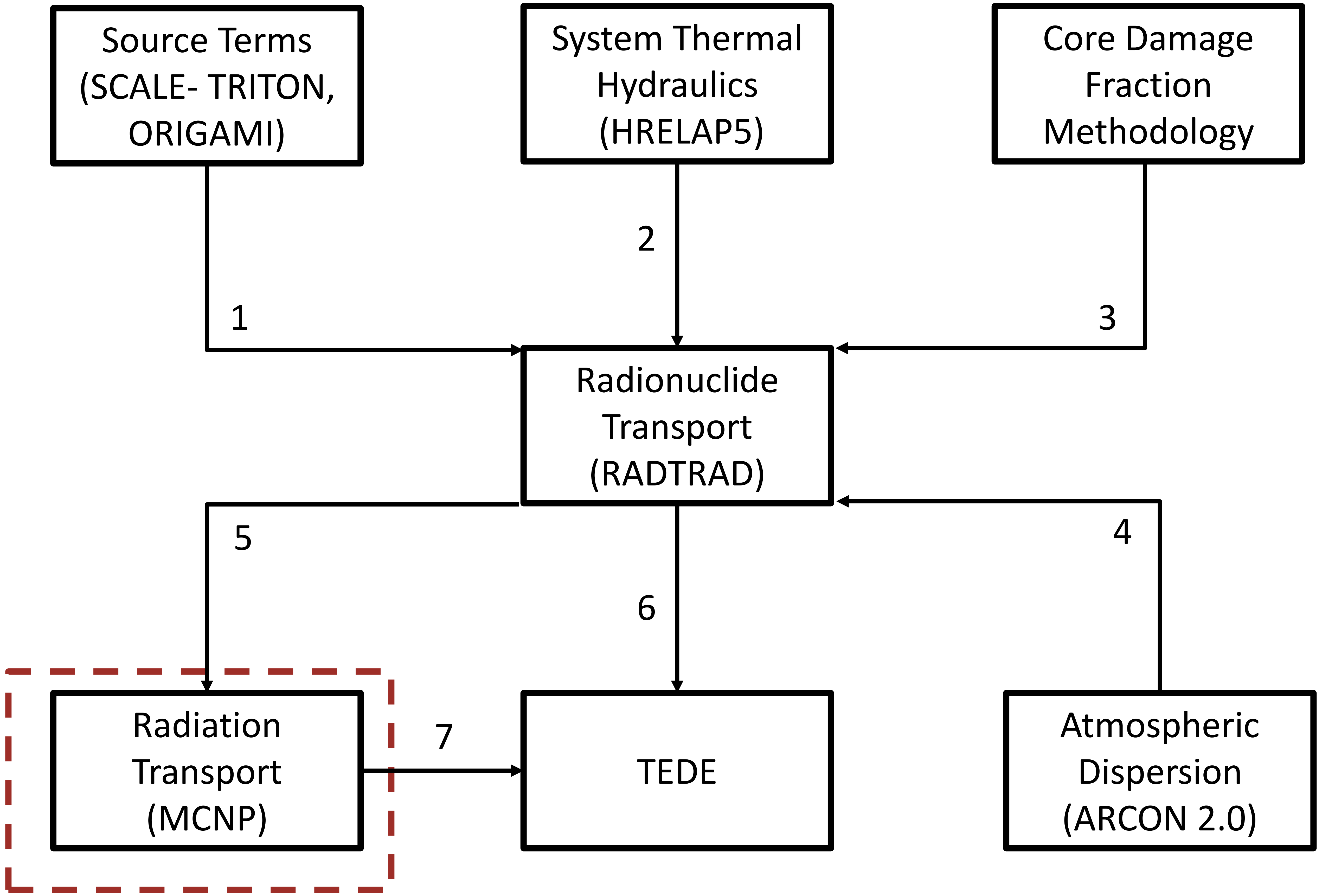
Atmospheric Dispersion: Source-to-Receptor Distances and Directions

- Distances and directions between source buildings and receptor points at EAB and LPZ are evaluated across 16 directional sectors (22.5° each) using 90° directional wind windows to determine 95th percentile atmospheric dispersion values.
  - Largest atmospheric dispersion factors among those for source-receptor combinations used in conformance with **Regulatory Position 2 of RG 1.194, Revision 0.**
  - MCR and TSC are known locations allowing for direct determination of source-receptor distance and direction



Example of the closest point of a building within each of the 16 directional sectors from the release points to the EAB/LPZ boundaries , in 45-degree windows

Analysis Flowchart: Radiation Transport



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Radiation Transport

- MCNP used to calculate radiation shine dose using RADTRAD provided radionuclide concentrations for:
  - External radioactive plume released from facility
  - Radioactive material in systems and components inside or external to control room (e.g., radioactive material buildup in filters)
  - Radioactive material in containment
- Bounding scenario with highest activity from each source used

Summary

- Methodology largely follows **RG 1.183 R1**
- Key unique features:
  - ✓ Use of **RG 1.183 R1** iodine chemical form fractions, justified based on NUREG/CR-5950 and bounding SMR-300 accident pH
  - ✓ Credit of partial flashing in steaming region OTSG
  - ✓ Use of sector-specific 95th percentile atmospheric dispersion coefficients for all accident dose analysis, justified based on onsite SMR-300 EAB and LPZ
- Not included in this methodology (to be covered in separate licensing actions):
  - ✓ Methodology to determine failed fuel fractions
  - ✓ Methodology for post-accident pH analysis
  - ✓ MHA LOCA assumptions to be used in EPZ sizing methodology

Open Session