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Sandia National Laboratories

Securing the future of Nuclear Energy

MELCOR

MELCOR Integrated Non-LWR Accident Progression and Source Term Analysis

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Nuclear Energy Safety Modeling and Analysis (8852)

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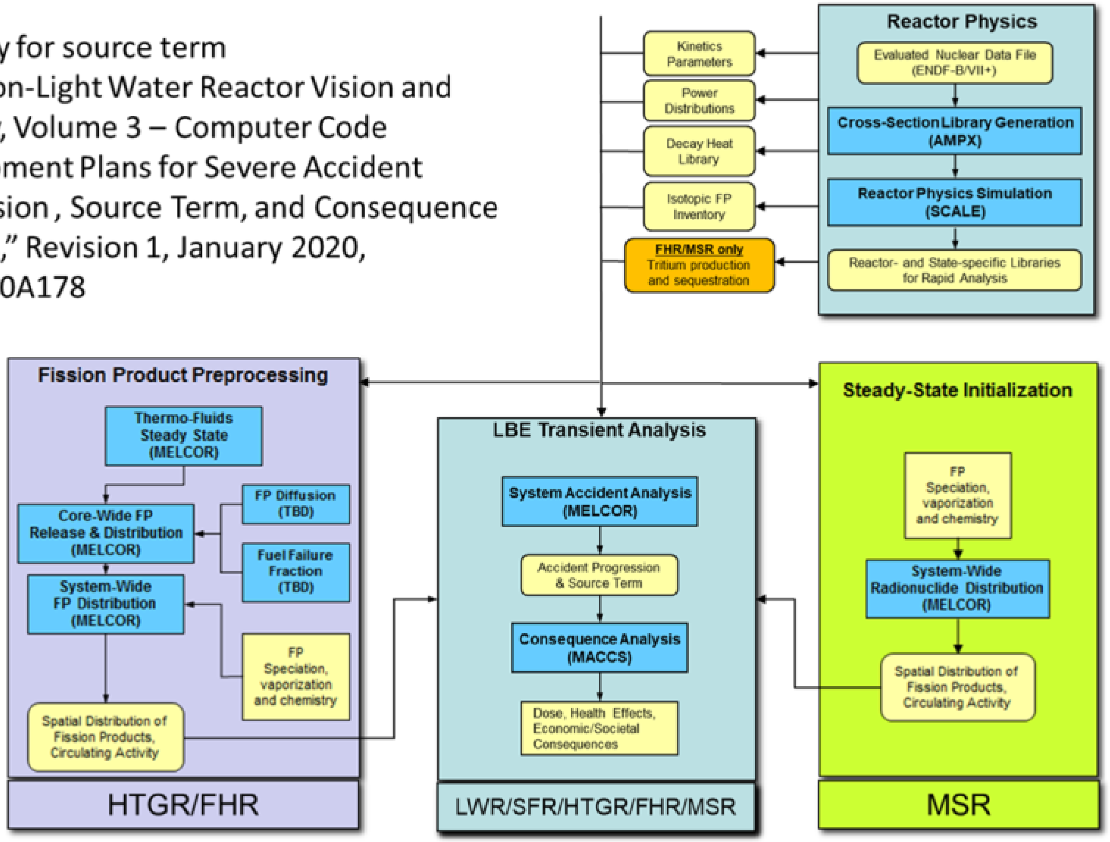
NRC strategy for severe accident analysis

Evaluation Model and Suite of Codes

Code strategy for source term

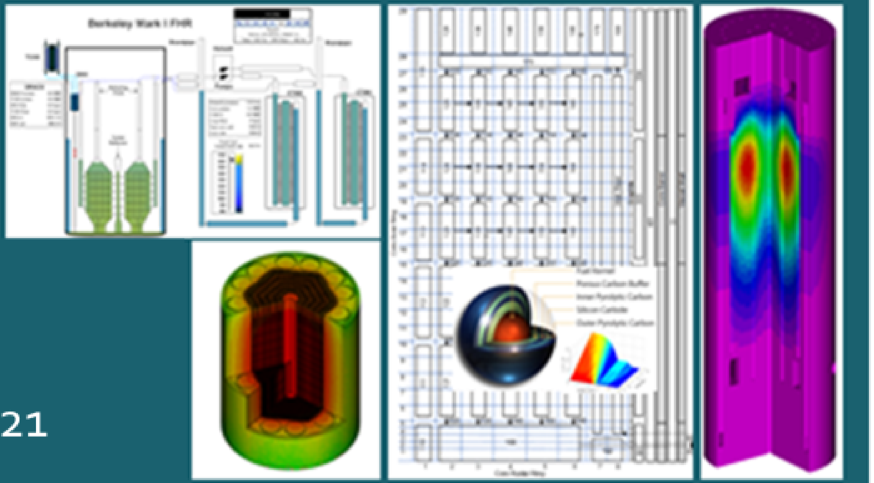
"NRC Non-Light Water Reactor Vision and Strategy, Volume 3 – Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis," Revision 1, January 2020, ML20030A178



MELCOR Non-LWR Public Demonstrations

Public Workshop: SCALE/ MELCOR Non LWR Source Term Demonstration Project

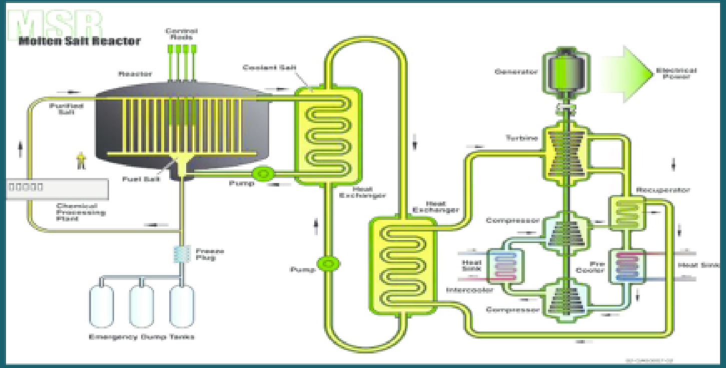
Heat pipe reactor – June 29, 2021
Gas cooled reactor – July 20, 2021
Pebble bed molten-salt-cooled reactor – Sept 14, 2021

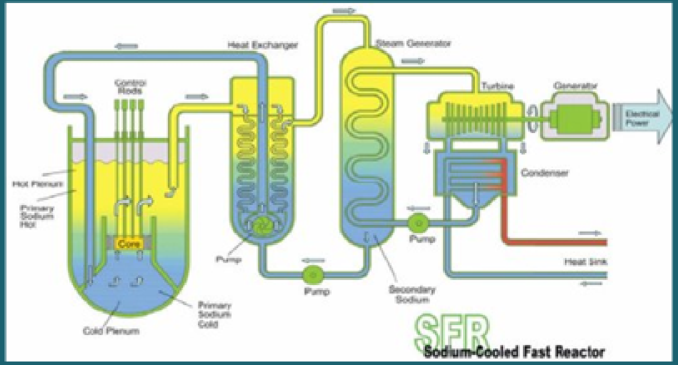


For More Information

Coming in 2022

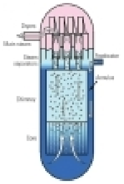
Molten-salt fueled reactor
Sodium-cooled fast reactor





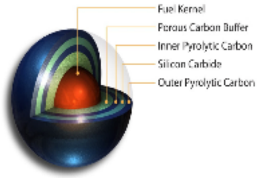
MELCOR – Integral Safety Analysis

Thermal hydraulics

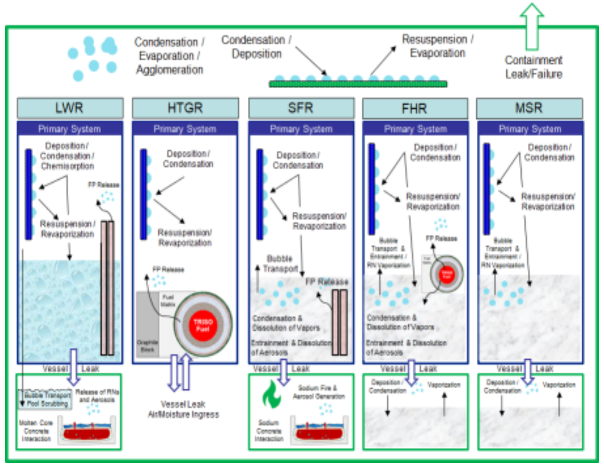


Neutronics

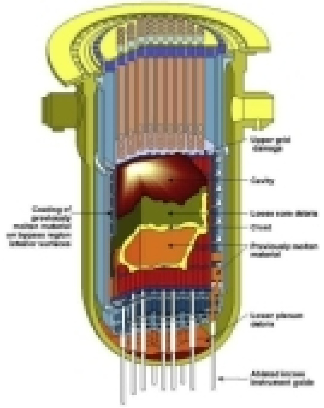
Fuel thermal-mechanical response



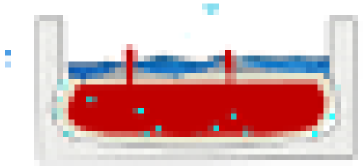
Fission product release and transport



Core degradation



Ex-vessel damage progression



MELCOR Pedigree – Leveraging Collaboration

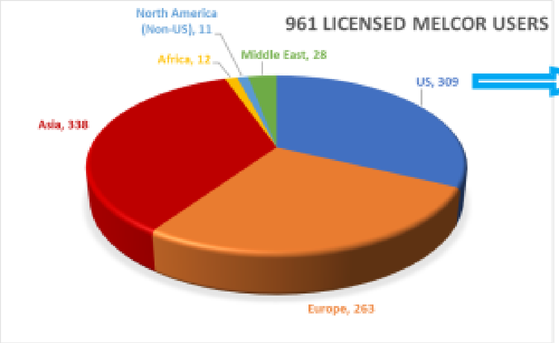
Validated physical models

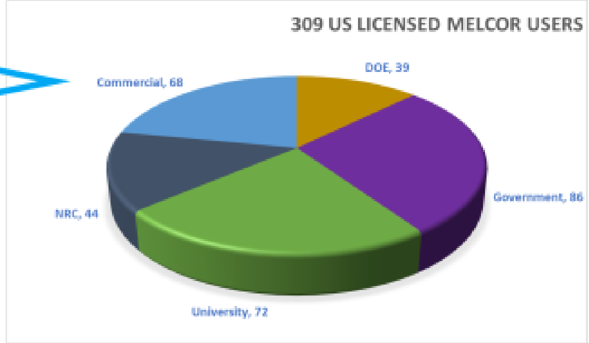
- International Standard Problems, benchmarks, experiments, and reactor accidents
- Beyond design basis validation will always be limited by model uncertainty that arises when extrapolated to reactor-scale

Cooperative Severe Accident Research Program (CSARP) is an NRC-sponsored international, collaborative community supporting the validation of MELCOR

International LWR fleet relies on safety assessments performed with the MELCOR code

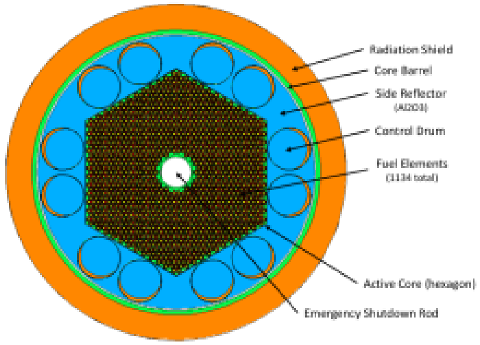
International Collaboration
Cooperative Severe Accident Research Program (CSARP) – June/U.S.A
MELCOR Code Assessment Program (MCAP) – June/U.S.A
European MELCOR User Group (EMUG) Meeting – Spring/Europe
European MELCOR User Group (EMUG) Meeting – Fall/Asia

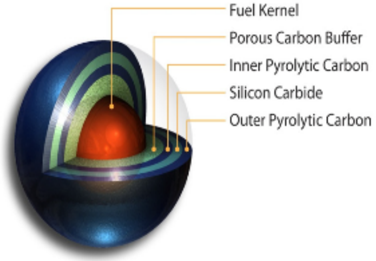


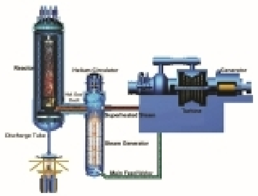


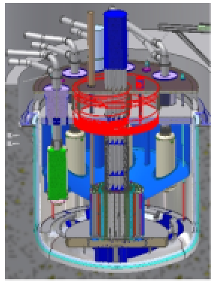
MELCOR Non-LWR Modeling

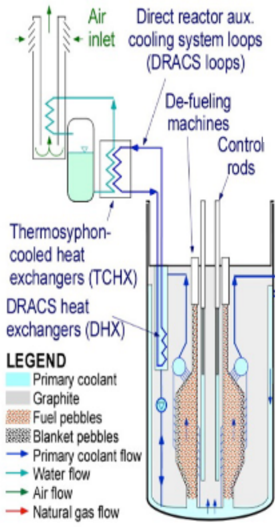
- Hydrodynamic modeling
  - Generalized working fluid treatment
  - *Conduction heat transfer within working fluids (under development)*
  - Generalized convection and flow models to capture flow through new core geometries (e.g., pebble beds)
- Core models
  - TRISO pebble and compact core components
  - Heat pipe reactor core component
  - Graphite oxidation
  - Intercell and intracell conduction
  - *Fast reactor core degradation (under development)*
- Fission product release
  - Generalized release modeling for metallic fuels
  - Radionuclide transport and release from TRISO particles, pebbles and compacts
  - *Generalized Radionuclide Transport and Retention (GRTR) model (under development)*
- Simplified neutronic modeling
  - Solid fuel core point kinetics
  - Fluid point kinetics (liquid-fueled molten salt reactors)

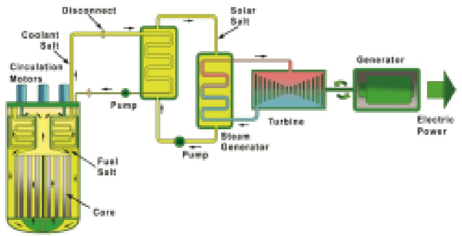






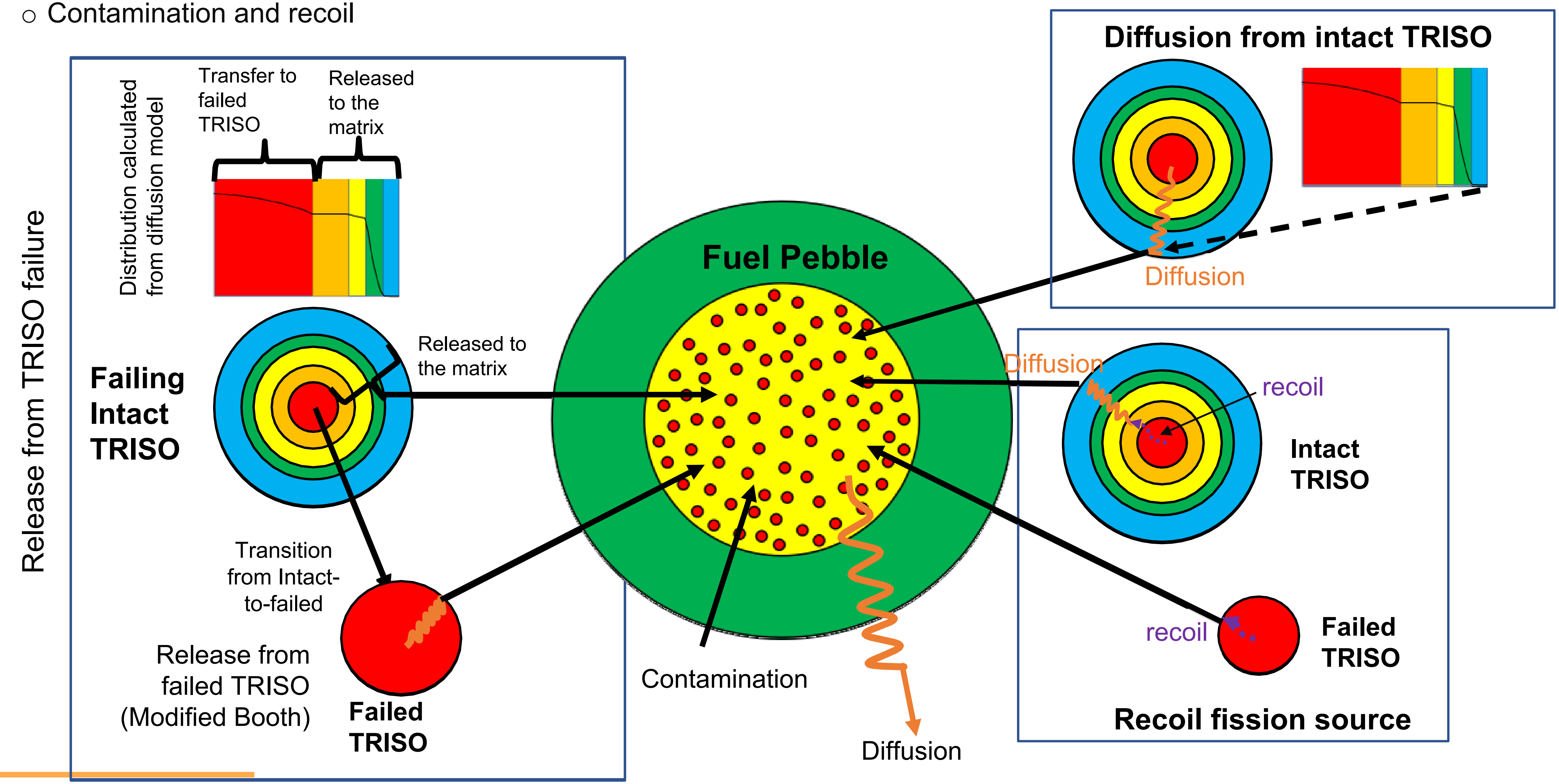






TRISO Radionuclide Release Models

- Recent failures – particles failing within latest time-step (burst release, diffusion release in time-step)
- Previous failures – particles failing on a previous time-step (time history of diffusion release)
- Contamination and recoil

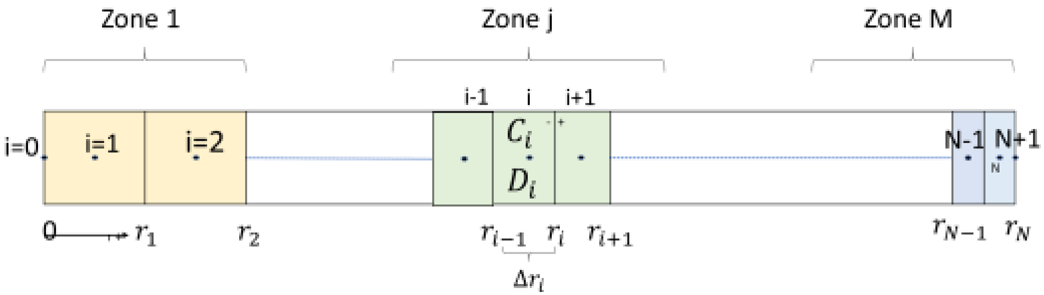


TRISO Radionuclide Diffusion Release Model

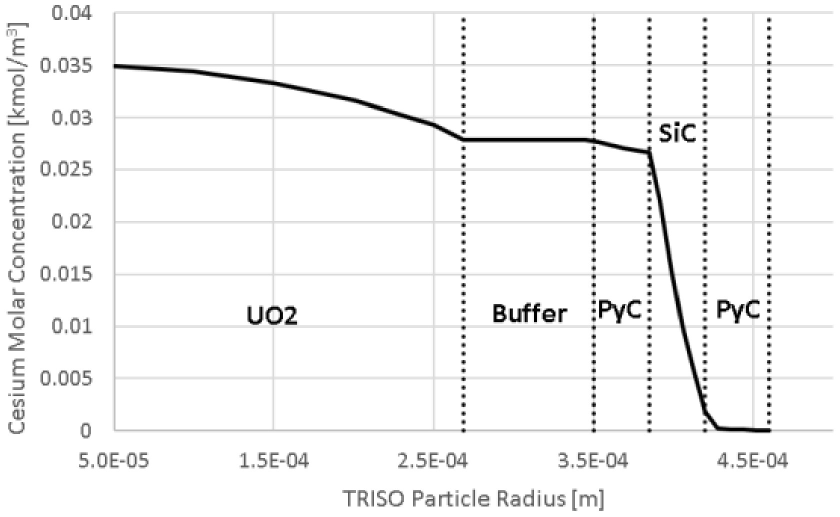
Intact TRISO Particles

- One-dimensional finite volume diffusion equation solver for multiple zones (materials)
- Temperature-dependent diffusion coefficients (Arrhenius form)

$$\frac{\partial C}{\partial t} = \frac{1}{r^n}\frac{\partial}{\partial r}\left(r^n D \frac{\partial C}{\partial r}\right) - \lambda C + \beta \qquad D(T) = D_0 e^{-\frac{Q}{RT}}$$



Intact TRISO Concentrations



Diffusivity Data Availability

Radionuclide	UO_2	UCO	PyC	Porous Carbon	SiC	Matrix Graphite	TRISO Overall
Ag	Some	Not investigated	Some	Not found	Extensive	Some	Extensive
Cs	Some		Some		Extensive	Some	Some
I	Some		Some		Some	Not found	Not found
Kr	Some		Some		Not found	Some	Some
Sr	Some		Some		Extensive	Some	Some
Xe	Some		Some		Some	Some	Not found

Data used in the demo calculation

[IAEA TECDOC-0978]

Layer	FP Species							
	Kr		Cs		Sr		Ag	
	D (m^2/s)	Q (J/mole)	D (m^2/s)	Q (J/mole)	D (m^2/s)	Q (J/mole)	D (m2/s)	Q (J/mole)
Kernel (normal)	1.3E-12	126000.0	5.6-8	209000.0	2.2E-3	488000.0	6.75E-9	165000.0
Buffer	1.0E-8	0.0	1.0E-8	0.0	1.0E-8	0.0	1.0E-8	0.0
PyC	2.9E-8	291000.0	6.3E-8	222000.0	2.3E-6	197000.0	5.3E-9	154000.0
SiC	3.7E+1	657000.0	7.2E-14	125000.0	1.25E-9	205000.0	3.6E-9	215000.0
Matrix Carbon	6.0E-6	0.0	3.6E-4	189000.0	1.0E-2	303000.0	1.6E00	258000.0
Str. Carbon	6.0E-6	0.0	1.7E-6	149000.0	1.7E-2	268000.0	1.6E00	258000.0

Iodine assumed to behave like Kr
CORSOR-Booth LWR scaling used to estimate other radionuclides

Heat Pipe Reactor Modeling

MELCOR

Release from fuel to reactor vessel

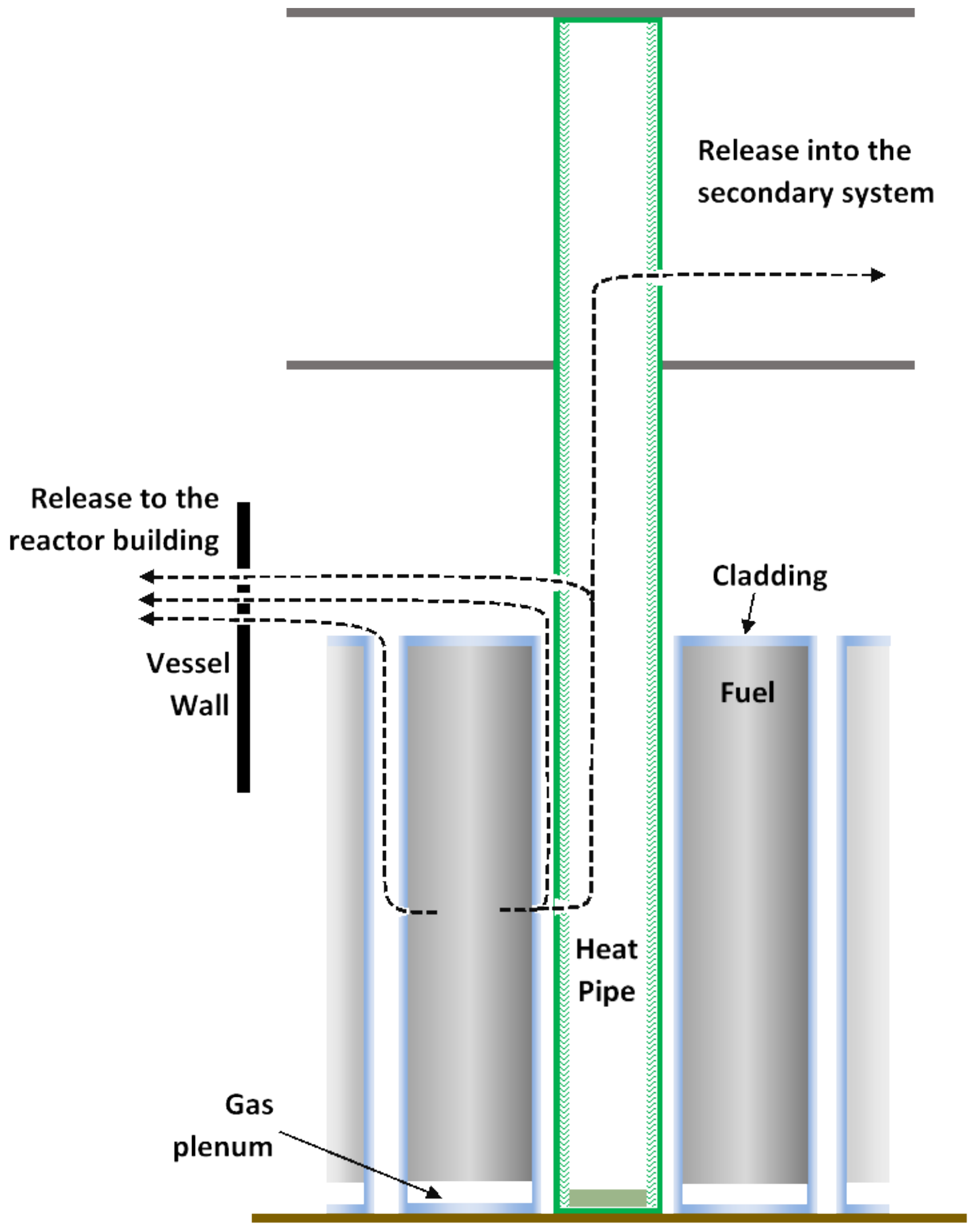
- Stainless-steel cladding failure at 1650 K

Release from reactor vessel to reactor building

- Assumed reactor vessel leakage

Heat-pipe release path

- Requires heat-pipe wall failure in two places
  - Creep rupture followed by melting
- Creep rupture failure in the heat-pipe condenser region (secondary system region) could lead to reactor building bypass



MELCOR Generalized Radionuclide Transport and Retention (GRTR) Model

Model Scope

Define 5 radionuclide physico-chemical forms in liquid pool
- Soluble fission products
- Insoluble fission products suspended in working fluid
- Insoluble fission products deposited on structures
- Insoluble fission products at liquid-gas interface
- Fission product gases

Generalized Gibbs Energy Minimization approach
- Fission product solubility
- Fission product vapor pressure

Model generically applies to range of non-LWR working fluids
- Molten salt systems
- Liquid metal systems

Fission Product Forms	Soluble FPs (salt-seeking) **Form 1**	Insoluble FP suspension **Form 2**	Salt liq./gas Interface **Form 3**	Insoluble FP deposits on structures **Form 4**	FPs Vapor/NCG **Form 5**
Equilibrium	Soluble & Volatile FP	Insoluble FP suspension			Gas Bubbles
Transport within GRTR	Soluble & Volatile FP	Insoluble FP suspension	Salt gas/liq. Interface	Deposits on heat structures; Deposits on core	Gas Bubble Rise
Gas space Release	Volatile Vapor and Eventual Aerosol	No Release	Bubble Film Rupture Aerosol	No Release	Gas Release (Kr, Xe, T)

MELCOR Generalized Radionuclide Transport and Retention (GRTR) State Modeling

Fission products characterized in terms of…

Isotopic state

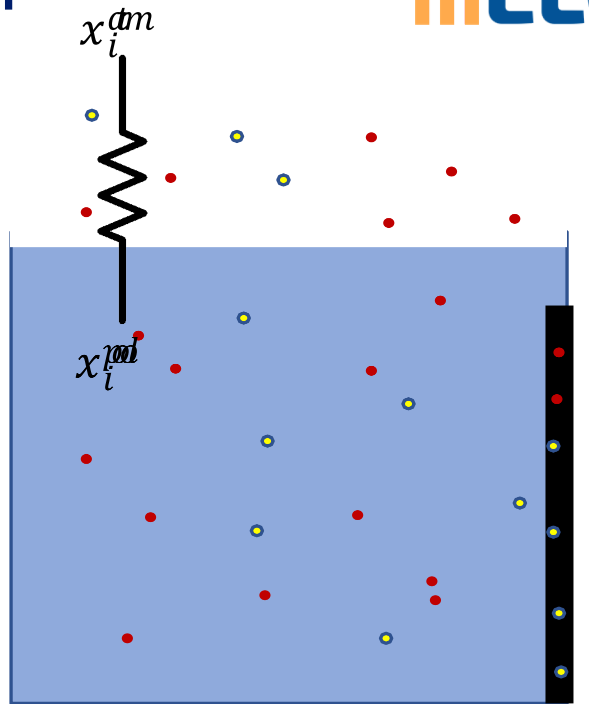
- Fission product decay

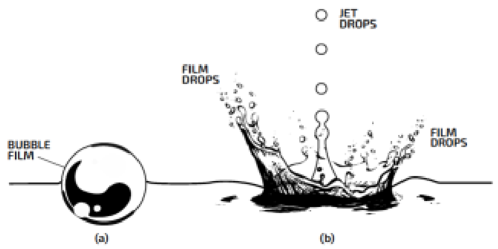
Distribution of fission products in reactor system

- Hydrodynamic flows moving fission products within system

Physico-chemical form and ability of fission products to be transported out of liquid pool

- Deposition on structures from liquid pool
- Vaporization into gas atmospheres from liquid pools
- Attachment to gas bubbles in liquid pool or surface interface between liquid pool and gas atmosphere
- Aerosolization of fission products into gas atmosphere through bursting of bubbles



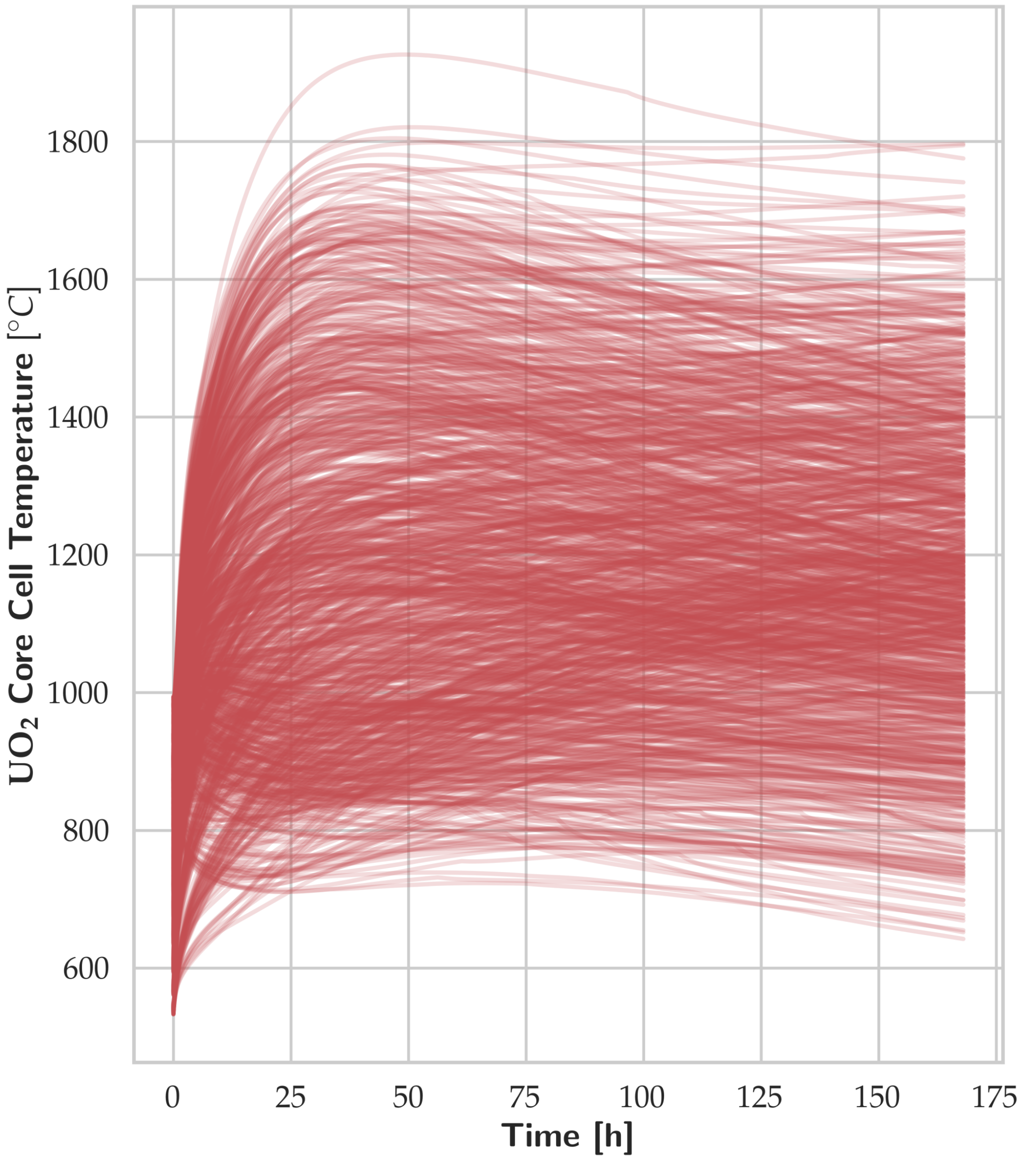


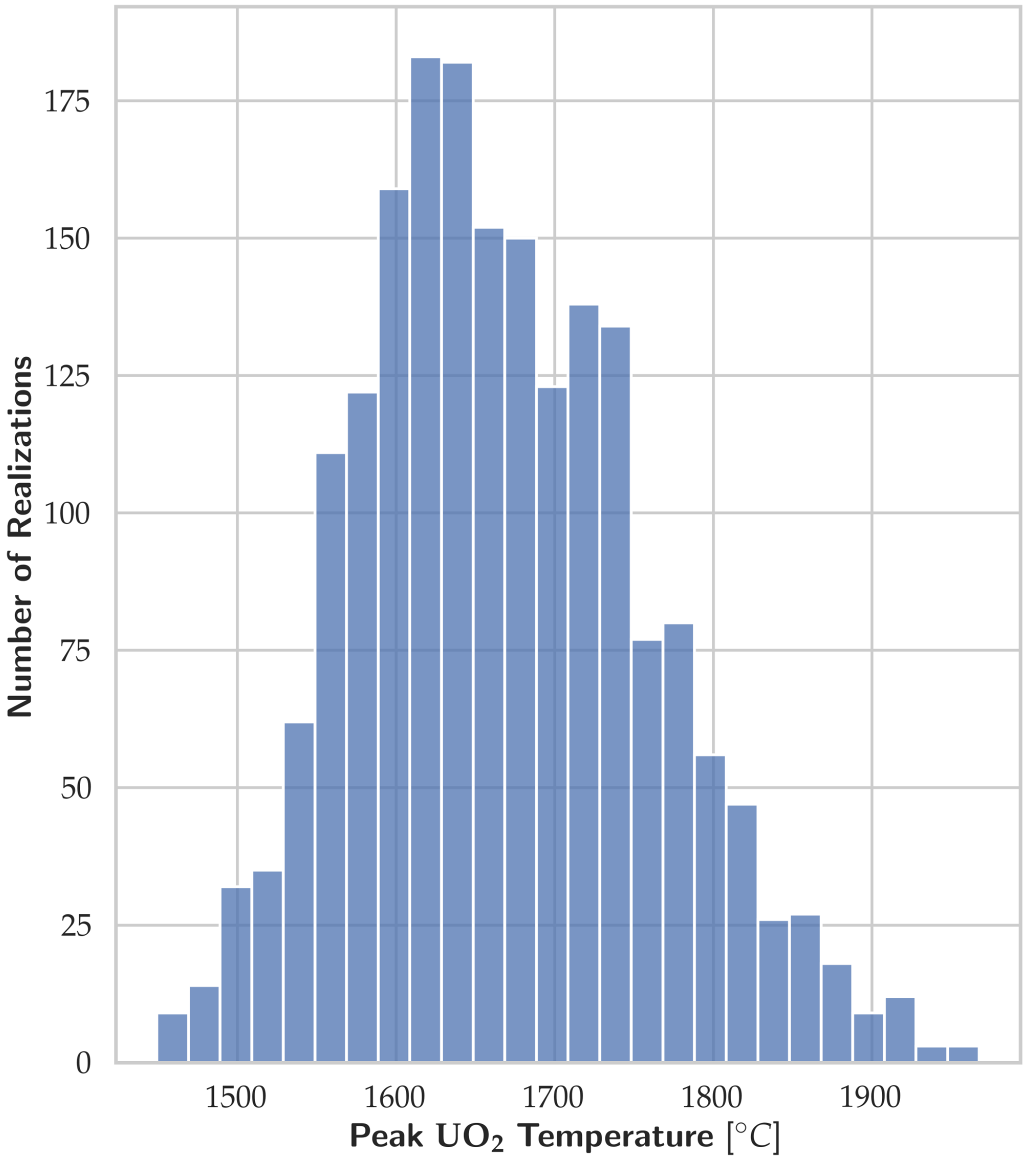
Note: MELCOR considers soluble, bulk colloid, interfacial colloid, and vapors as distinct physico-chemical states

MELCOR can be used to explore the variability of the results to uncertainties

Model	Parameter	Distribution	Range
TRISO Model Parameters	Initial TRISO Failure Fraction (fraction of inventory)	Log uniform	10^{-5} – 10^{-3}
	TRISO Failure Rate Multiplier (-)	Log uniform	0.1 – 10.0
	Intact TRISO Diffusivity Multiplier (-)	Log uniform	0.001 – 1000.0
	Failed TRISO Diffusivity Multiplier (-)	Log uniform	0.001 – 1000.0
	Matrix Diffusivity Multiplier (-)	Log uniform	0.001 – 1000.0
	TRISO Pebble Emissivity (-)	Uniform	0.5 – 0.999
	TRISO Pebble Bed Porosity (-)	Uniform	0.3 – 0.5
	TRISO recoil fraction (-)	Uniform	0 – 0.03
Radionuclide Model Parameters	Shape Factor (-)	Uniform	1.0 – 5.0
	Gaseous Iodine Multiplier (Base = 5% I_2)	Uniform	0.02 – 1.0
Design Parameters	Graphite Conductivity Multiplier (-)	Uniform	0.5 – 1.5
	Decay Heat Multiplier (-)	Uniform	0.9 – 1.1
	RCCS Blockage Multiplier (-)	Log uniform	0.001 – 1.0
	RCCS Emissivity (-)	Uniform	0.1 – 1.0
	Reactor Building Leakage Multiplier (-)	Log uniform	0.1 – 100.0
	Wind speed (m/s)	Uniform	0 - 10

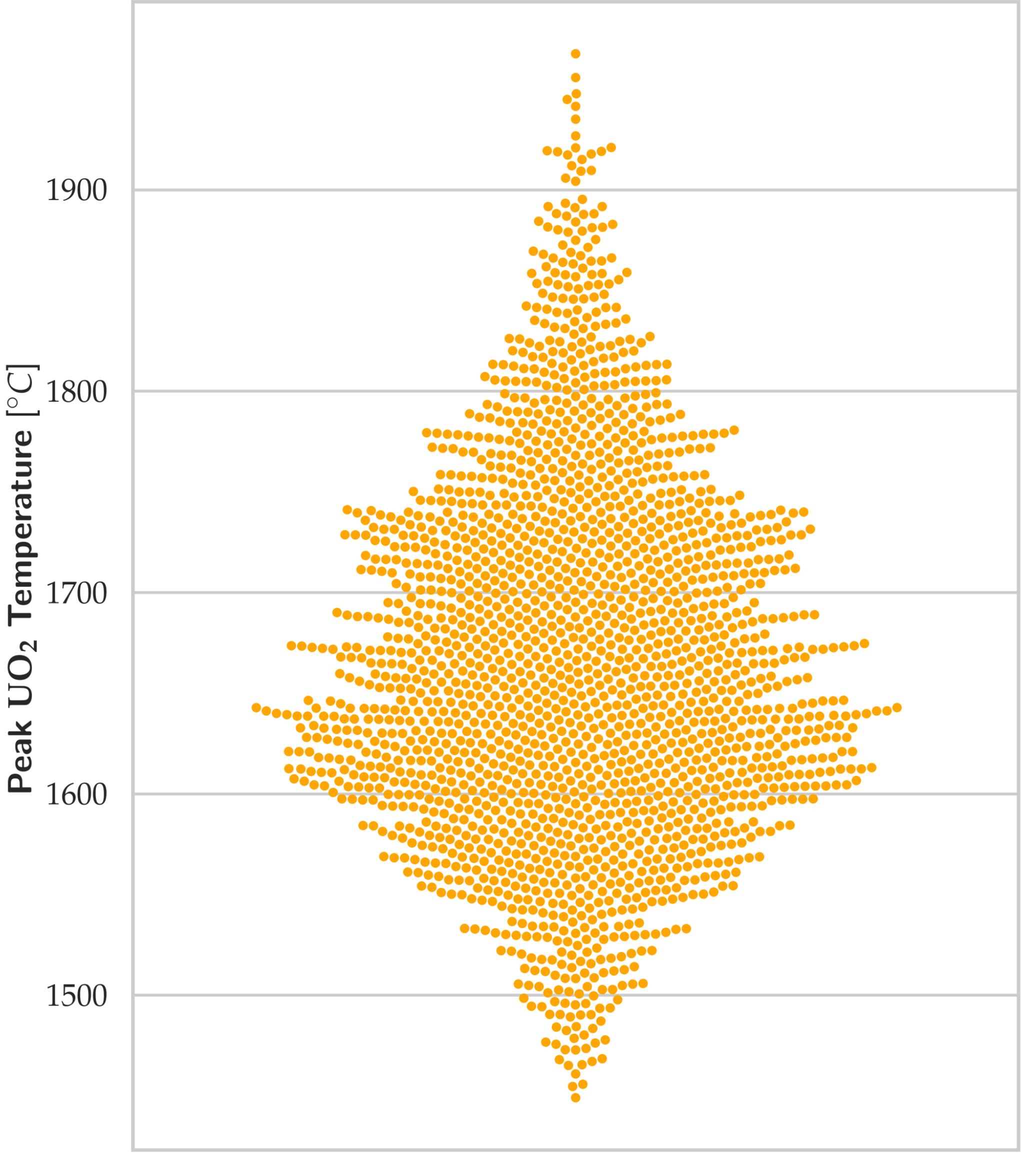
UO_2 Thermal Response

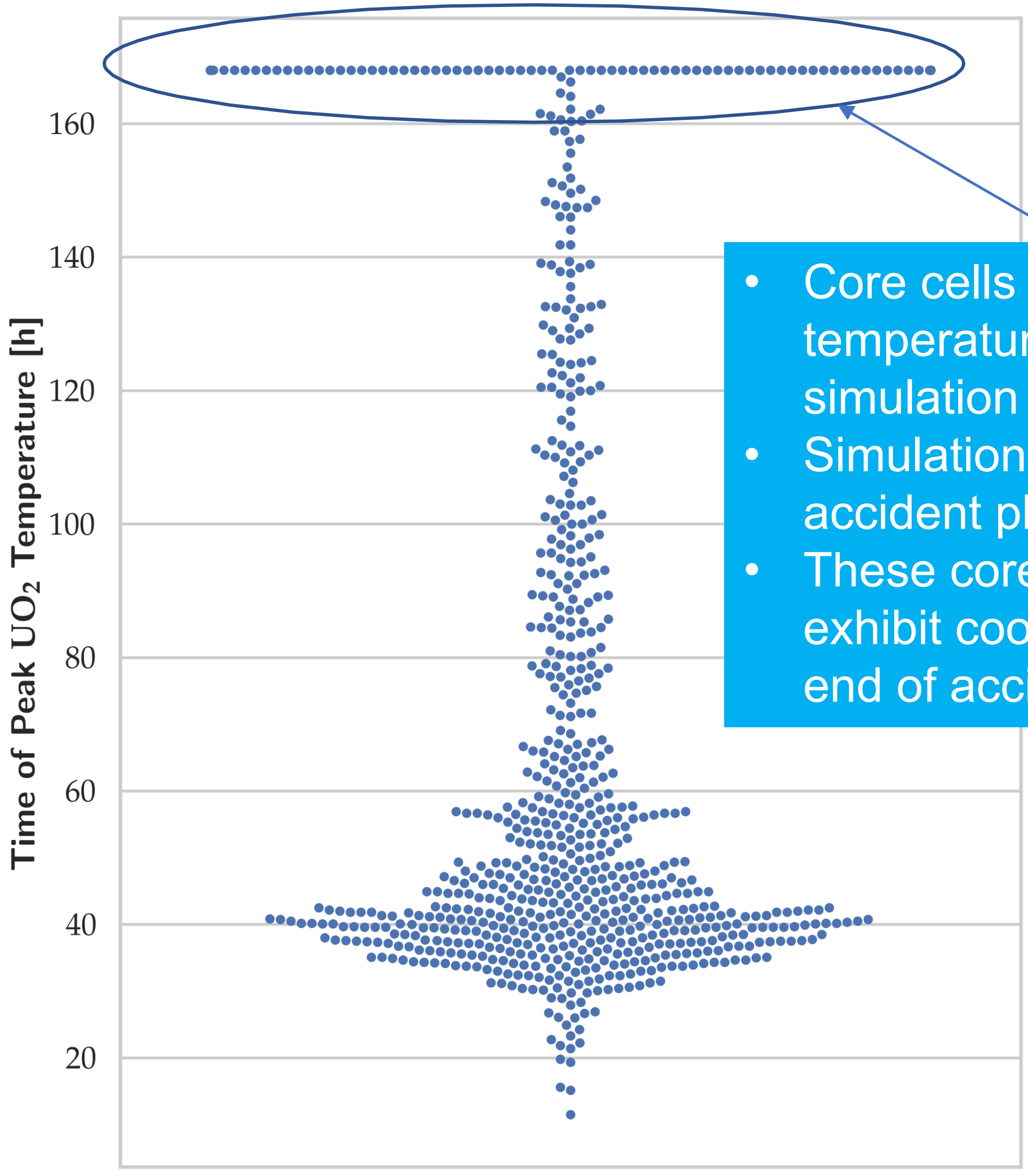




UO_2 Thermal Transient Evolution

ELCOR

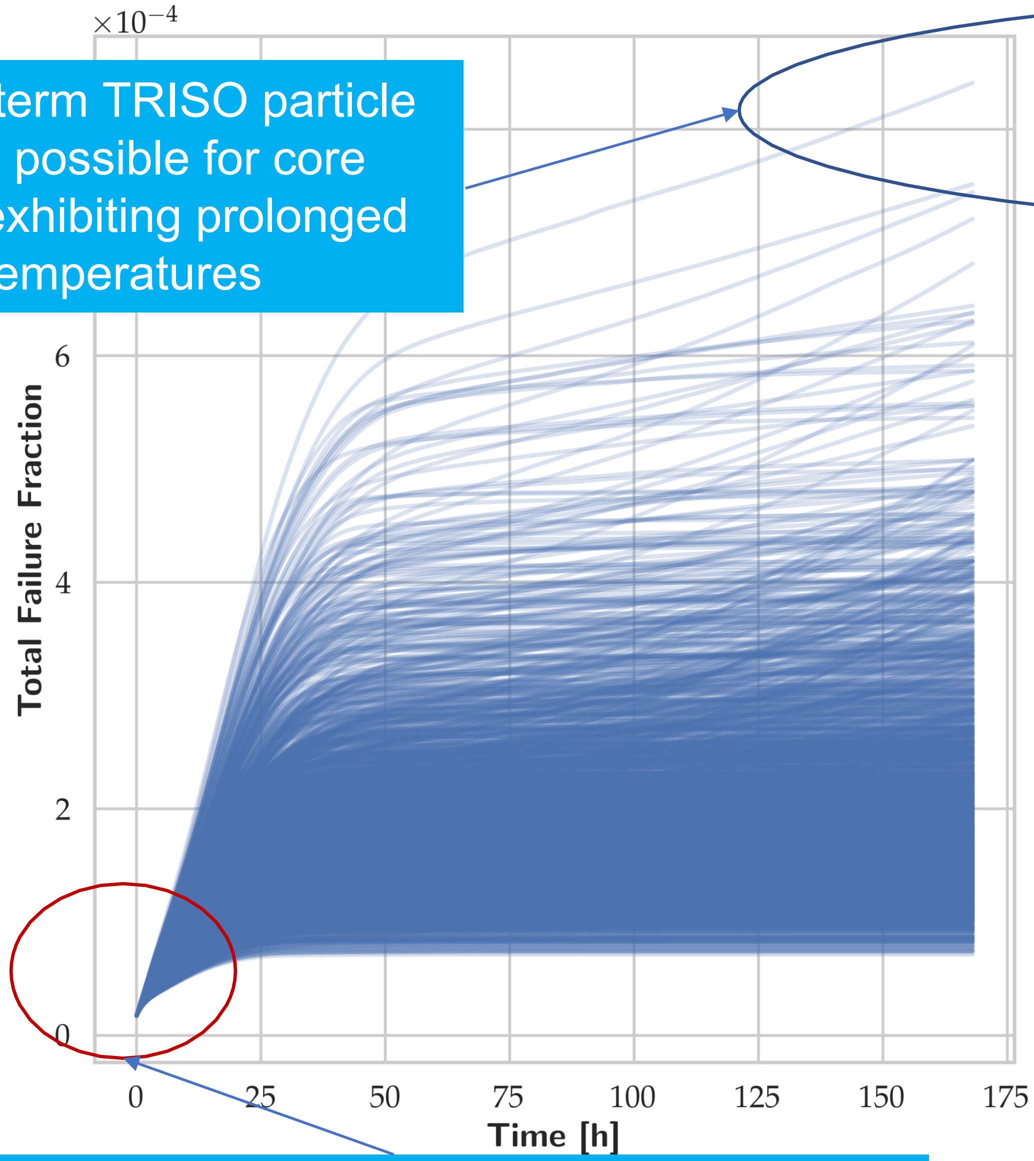


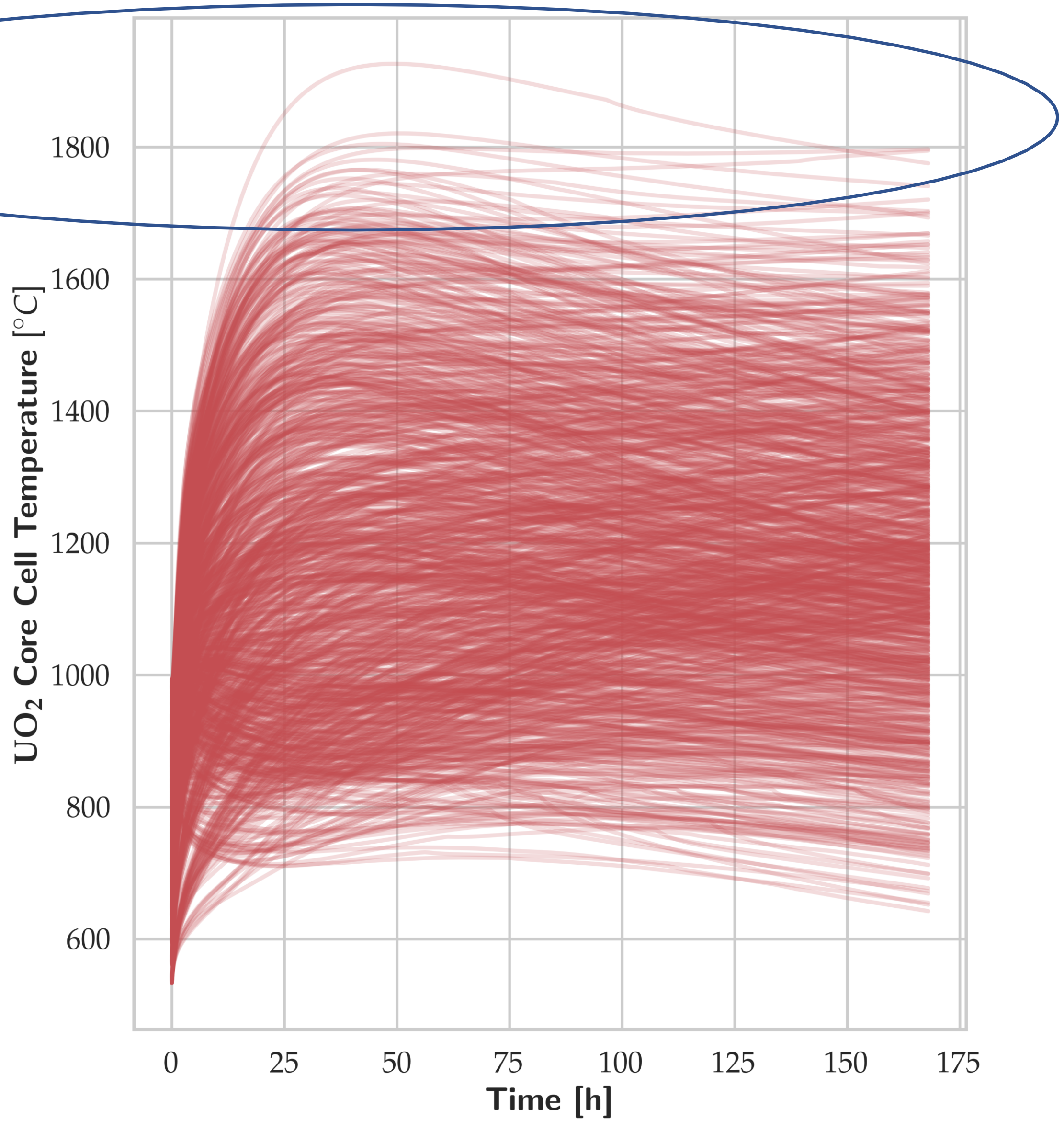


- Core cells with peak fuel temperatures at end of simulation
- Simulation time denoted as accident phase
- These core cells do not exhibit cooldown prior to end of accident phase

TRISO Particle Failure

Long-term TRISO particle failure possible for core cells exhibiting prolonged over-temperatures



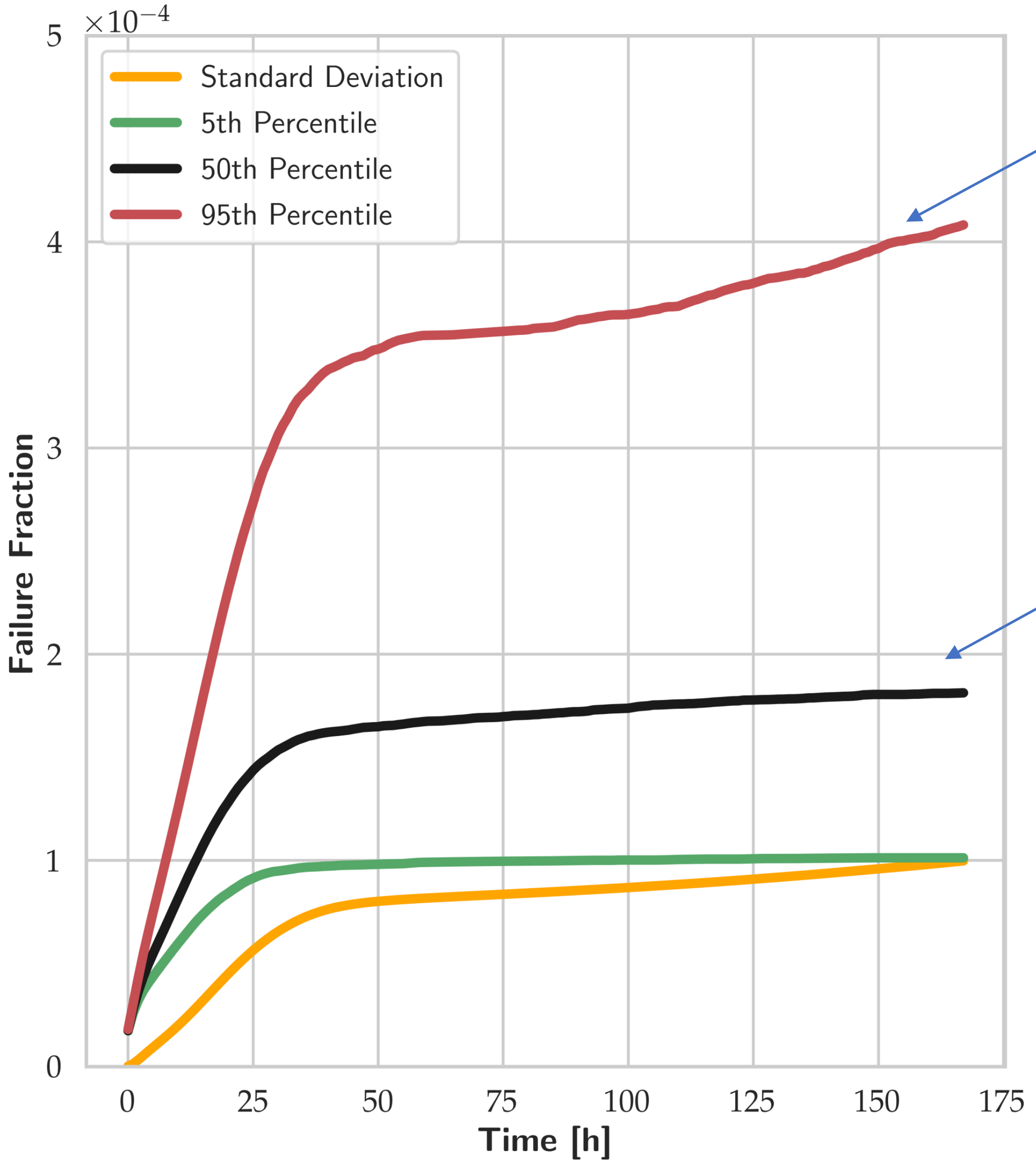


Initial distribution of failed TRISO particles

Evolution of TRISO Particle Failures

Long-term failures of TRISO particles at lower rate but driven by prolonged period of elevated fuel temperature

Rapid growth in failure fraction driven by the early temperature excursion



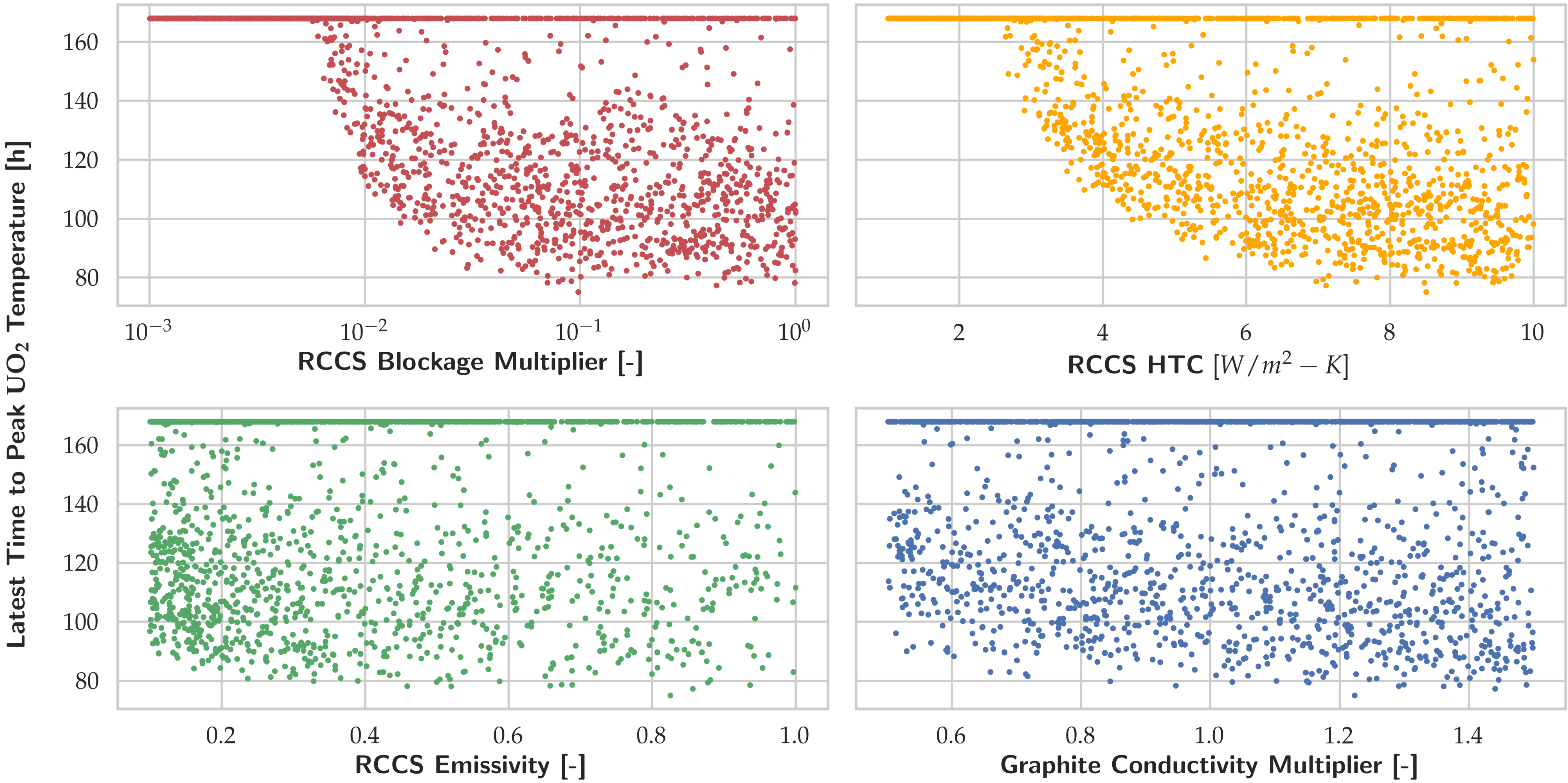
Tails of realizations contributing to longer term growth of TRISO particle failures

50th percentile reasonably stable in the long-term

Lower rates of failure entirely driven by early temperature excursion
Variability in peak fuel temperature and cooldown transient dominates higher failure rate realizations

Role of Decay Heat Rejection – Latest Time to Peak Fuel Temperature

ELCOR



Role of Decay Heat Rejection – Peak Fuel Temperature