SCALE/MELCOR Non-LWR Source Term Demonstration Project – Heat Pipe Reactor

June 2021

Outline

NRC strategy for non-LWR source term analysis

Project scope

Heat pipe reactor fission product inventory/decay heat methods and results

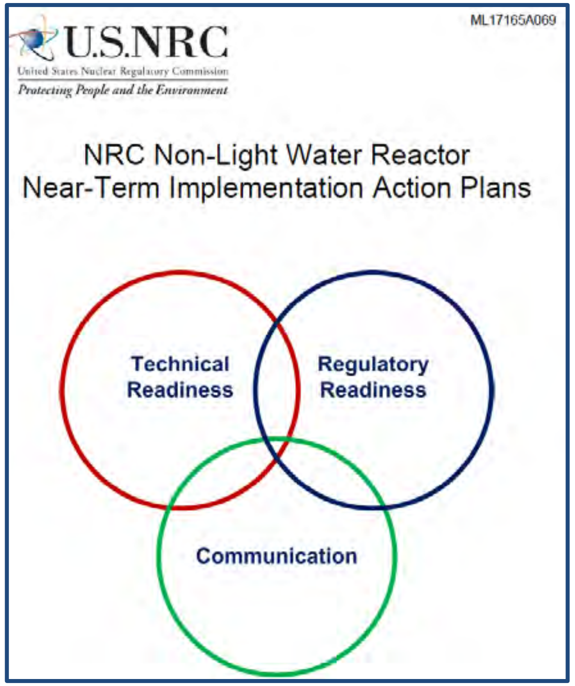
Heat pipe reactor plant model and source term analysis

Summary

Appendices

- SCALE overview
- MELCOR overview

Integrated Action Plan (IAP) for Advanced Reactors


ML17165A069

U.S.NRC

United States Nuclear Regulatory Commission

Protecting People and the Environment

NRC Non-Light Water Reactor
Near-Term Implementation Action Plans

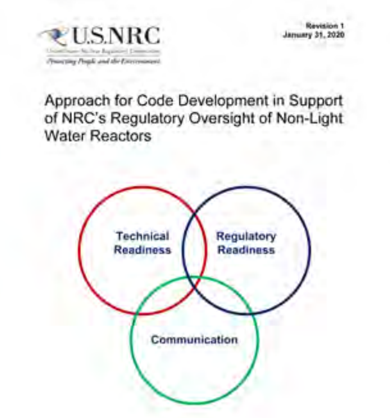
ML17165A069

Near-Term Implementation Action Plan

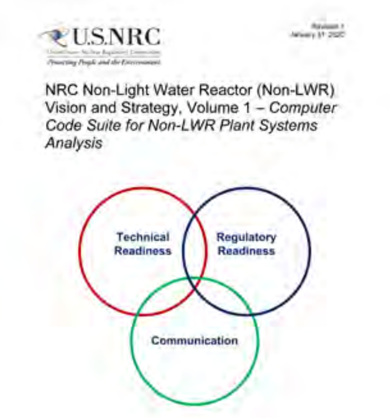
- Strategy 1
  Knowledge, Skills, and Capacity
- Strategy 2
  Analytical Tools
- Strategy 3
  Flexible Review Process
- Strategy 4
  Industry Codes and Standards
- Strategy 5
  Technology Inclusive Issues
- Strategy 6
  Communication

IAP Strategy 2 Volumes

These Volumes outline the specific analytical tools to enable independent analysis of non-LWRs, "gaps" in code capabilities and data, V&V needs and code development tasks.



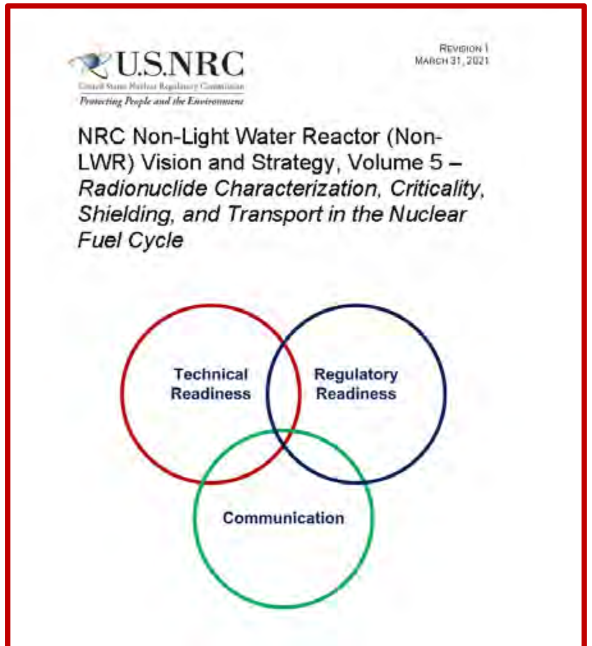
Introduction
ML20030A174



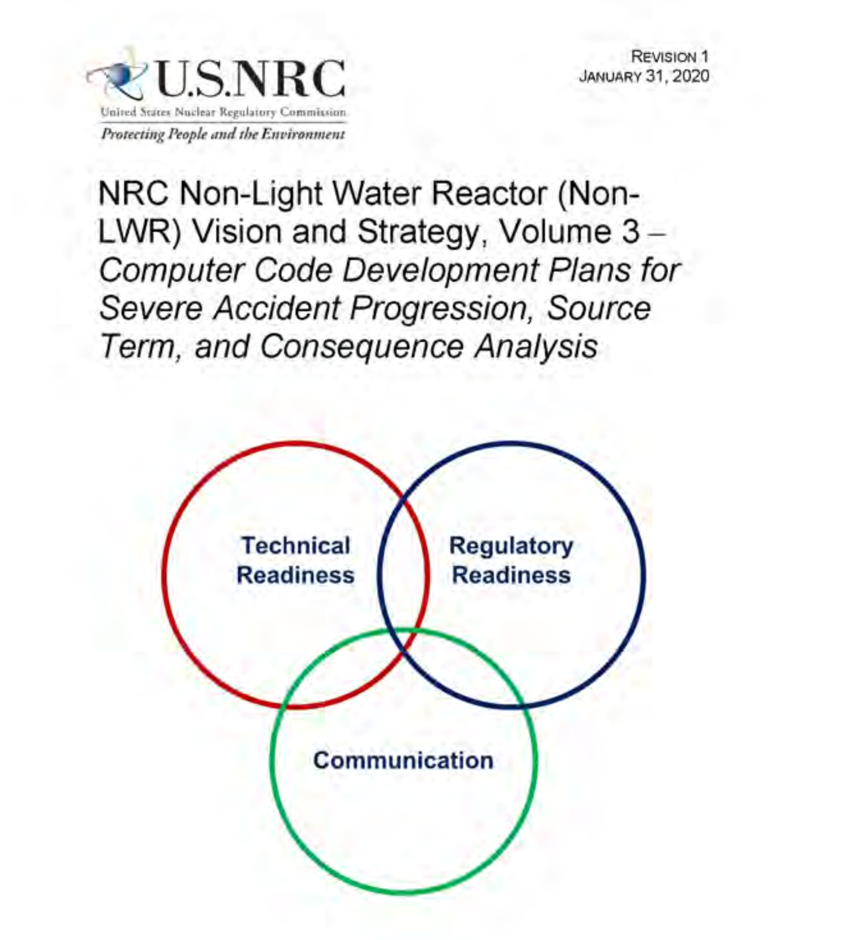
Volume 1
ML20030A176

Volume 2
ML20030A177

Volume 4
ML21085A484



Volume 5
ML21088A047



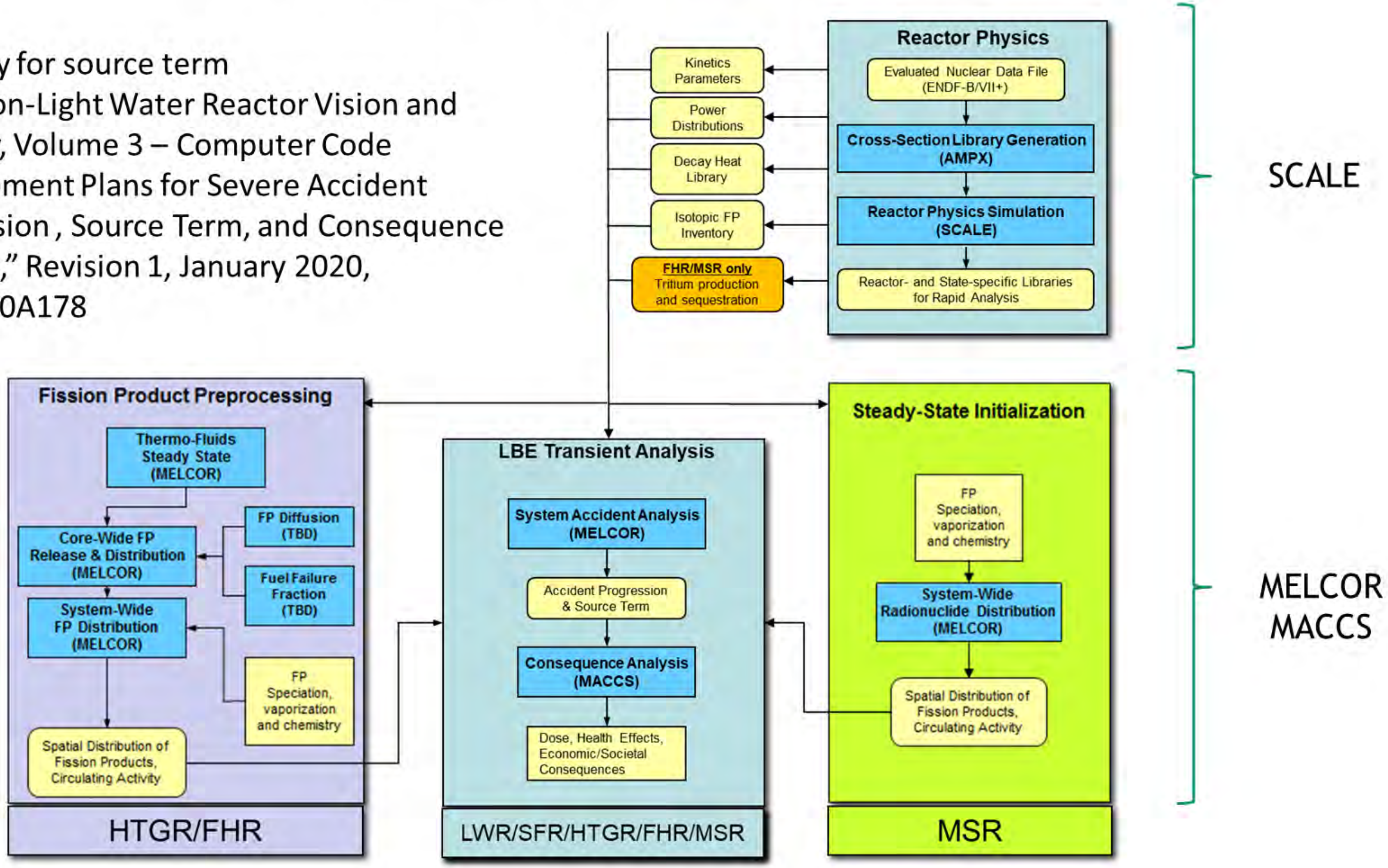
Volume 3
ML20030A178

NRC strategy for non-LWR analysis (Volume 3)

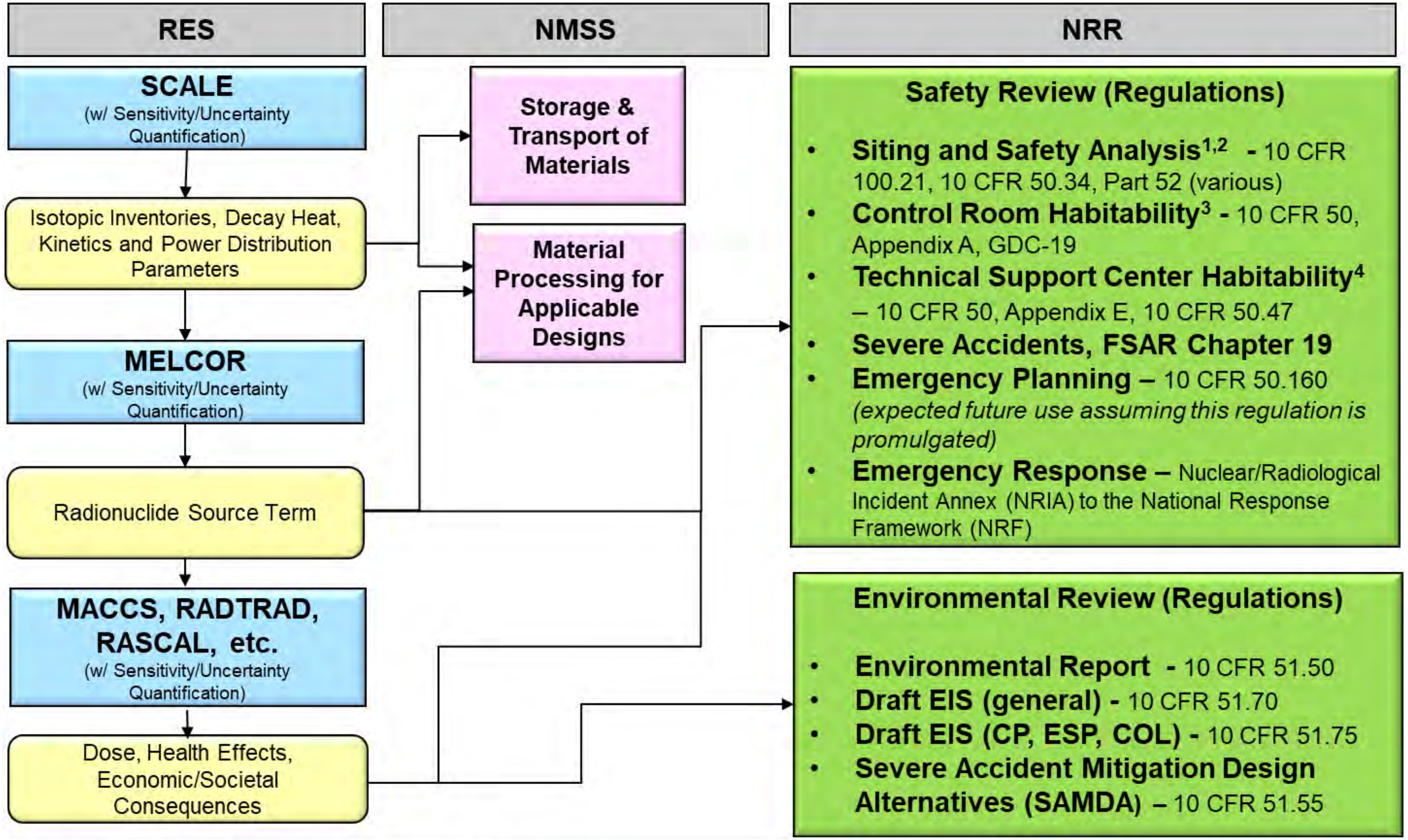
Evaluation Model and Suite of Codes

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



Role of NRC severe accident codes



Dose Criteria Reference Values (10 CFR 50/52)

1) An individual located at any point on the boundary of the **exclusion area** for any 2-hour period following the onset of postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv **(25 rem)** total effective dose equivalent (TEDE)
2) An individual located at any point on the outer boundary of the **low population zone**, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv **(25 rem)** TEDE
3) Adequate radiation protection is provided to permit access to and occupancy of the **control room** under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv **(5 rem)** TEDE
4) Dose criterion not in regulation but found in NUREG-0737/NUREG-0696. GDCs are applicable to light-water reactors. Non-LWRs will have principal design criteria (PDCs) which may have a similar requirement.

Project Scope

Project objectives

Understand severe accident behavior

- Provide insights for regulatory guidance

Facilitate dialogue on staff's approach for source term

Demonstrate use of SCALE and MELCOR

- Identify accident characteristics and uncertainties affecting source term
- Develop publicly available input models for representative designs

Project scope

Full-plant models for three representative non-LWRs (FY21)

- Heat pipe reactor – INL Design A
- Pebble-bed gas-cooled reactor – PBMR-400
- Pebble-bed molten-salt-cooled – UC Berkeley Mark I

FY22

- Molten-salt-fueled reactor – MSRE
- Sodium-cooled fast reactor – TBD

Project approach

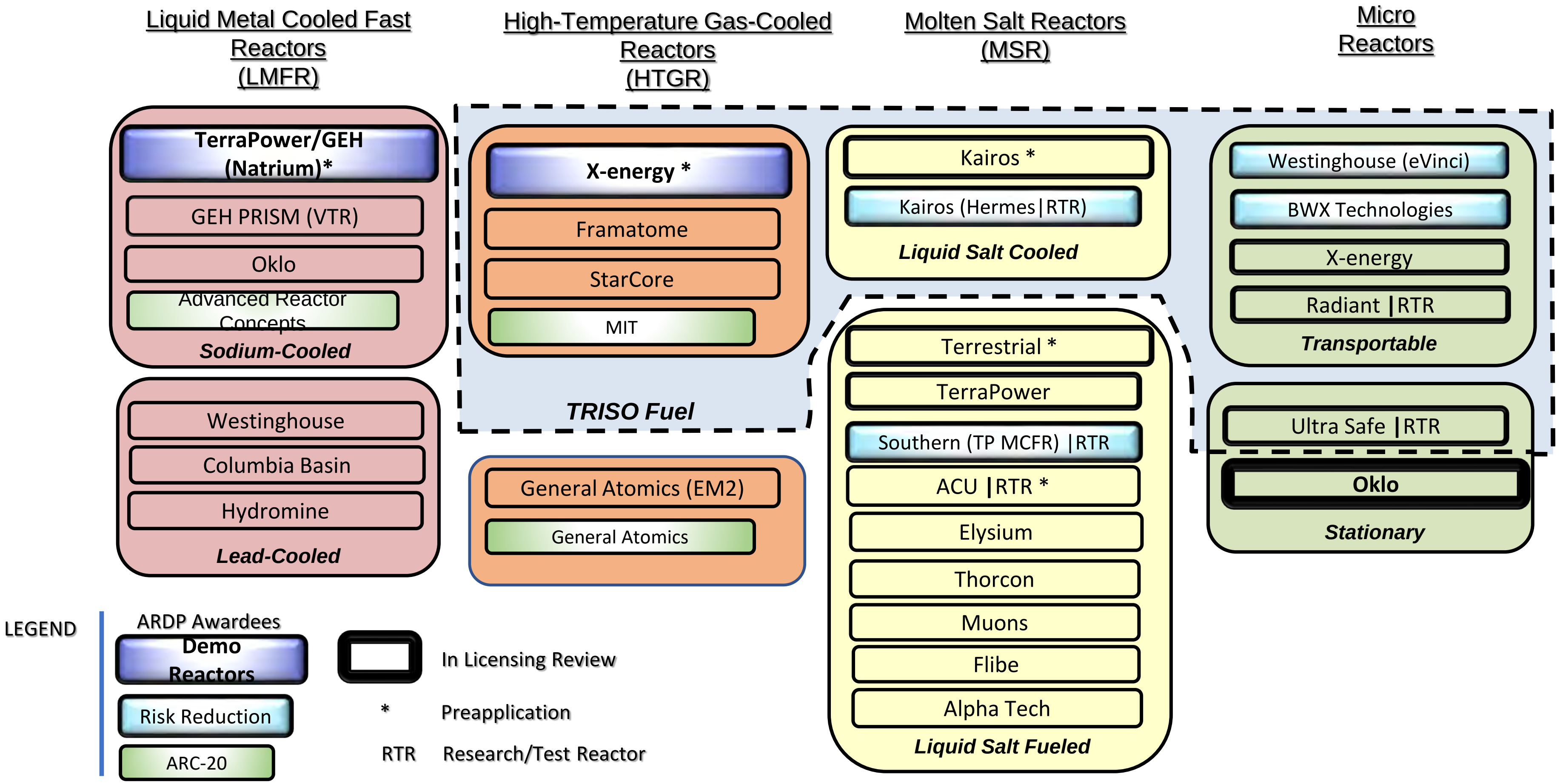
1. Develop SCALE model to provide MELCOR with decay heat, core radionuclide inventories, kinetics parameters, power distribution

2. Build MELCOR full-plant input model

3. Scenario selection

4. Perform simulations for the selected scenario and debug
   - Base case
   - Sensitivity cases
   - Uncertainty cases

Advanced Reactor Designs



Heat Pipe Reactor

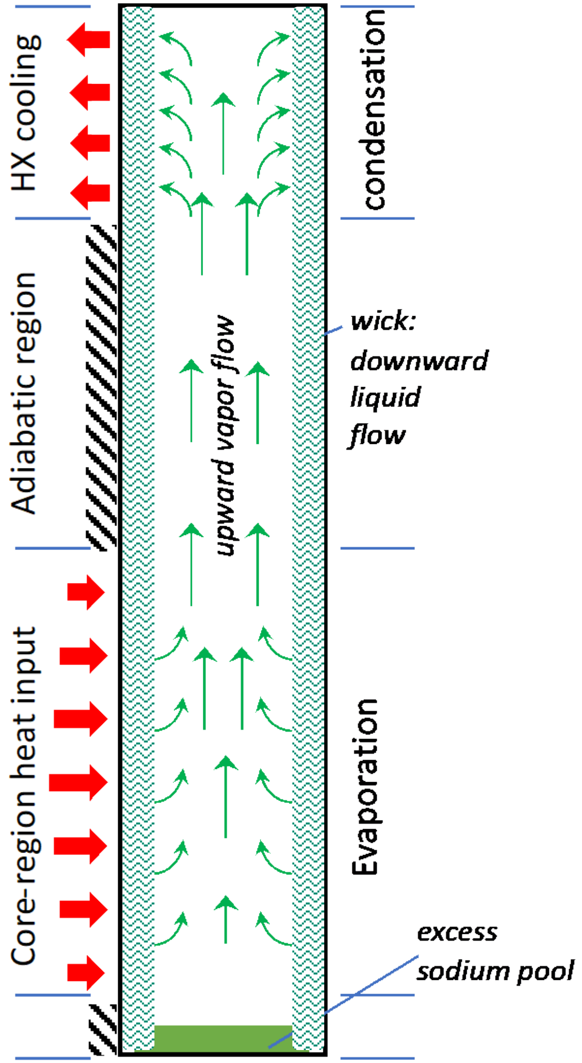
Heat pipe for reactor use

Construction

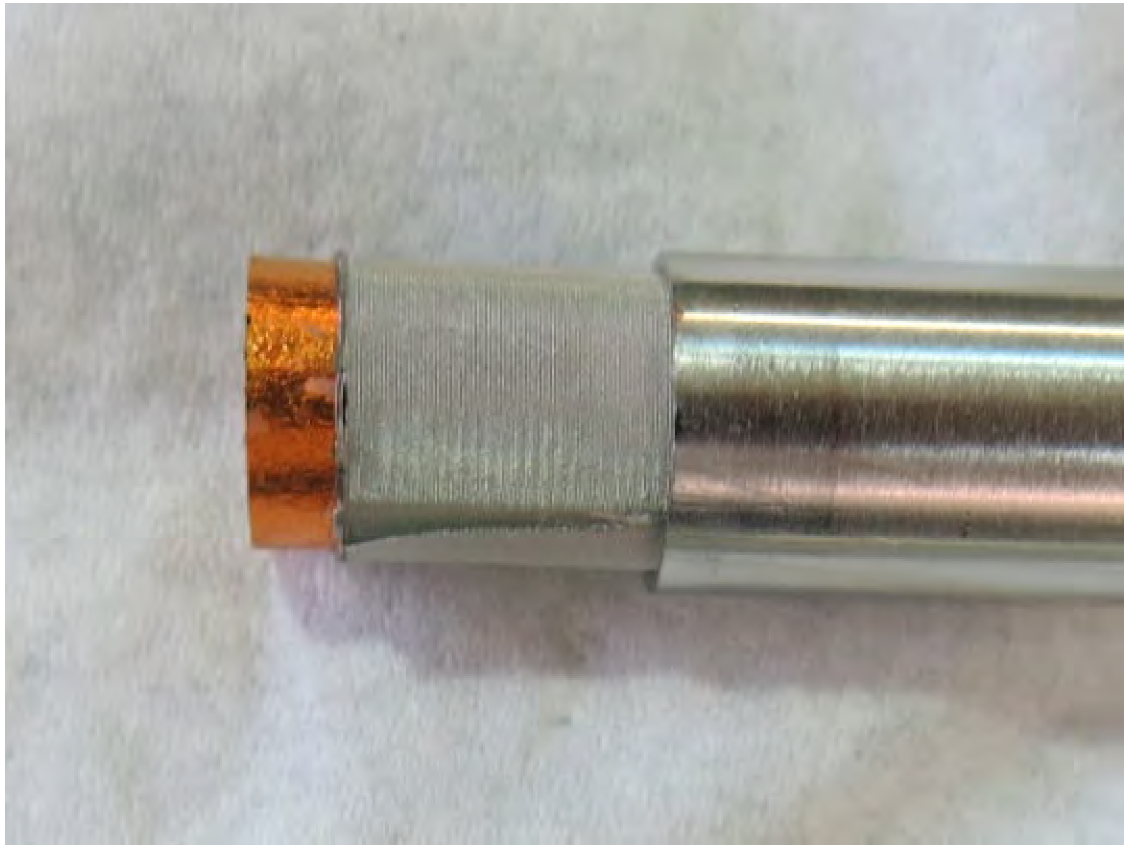
- Metal pipe with wick along pipe inside surface
- Liquid coolant fills area between wick and pipe inside surface

Operation

- The core heats the liquid coolant which generates vapor
- The vapor flows to the other end of the heat pipe where it condenses, heating the secondary system fluid
- Coolant film return flow by capillary forces



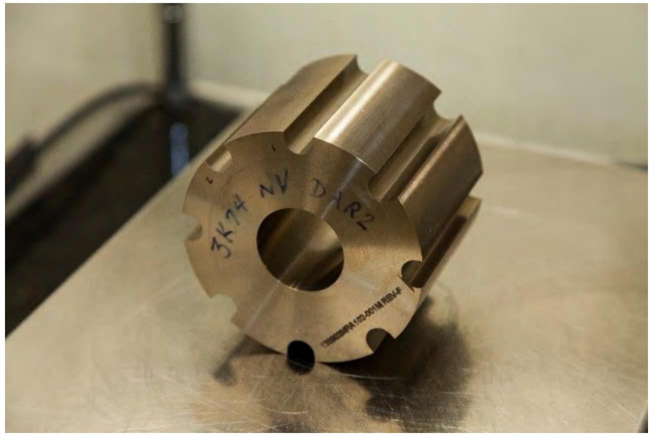
Heat pipe wick being installed



First heat pipe reactor

KRUSTY experiment

- Kilowatt Reactor Using Stirling TechnologY
- Part of NASA’s Kilopower project
- 3 kW thermal power
- 8 heat pipes clamped to uranium cylinder
- Heat pipes transfer heat to Sterling engine
- Operated 28 hours (March 20-21, 2018)
- LANL video on Kilopower



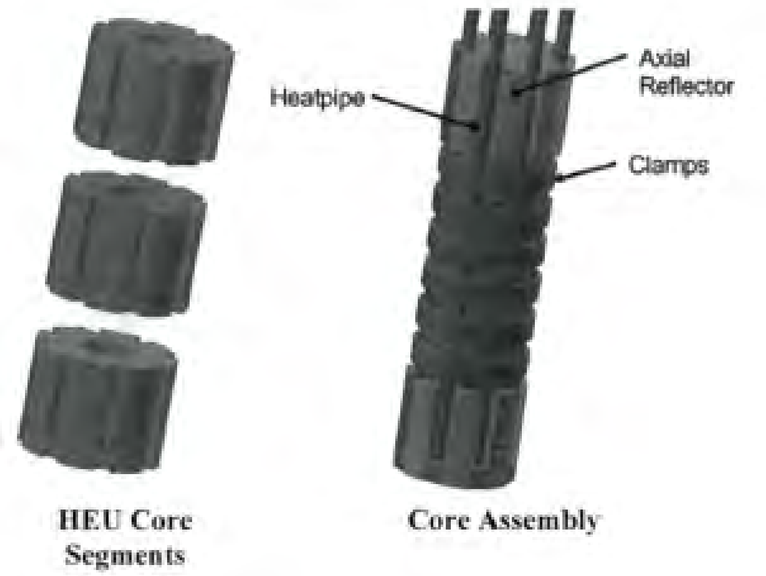
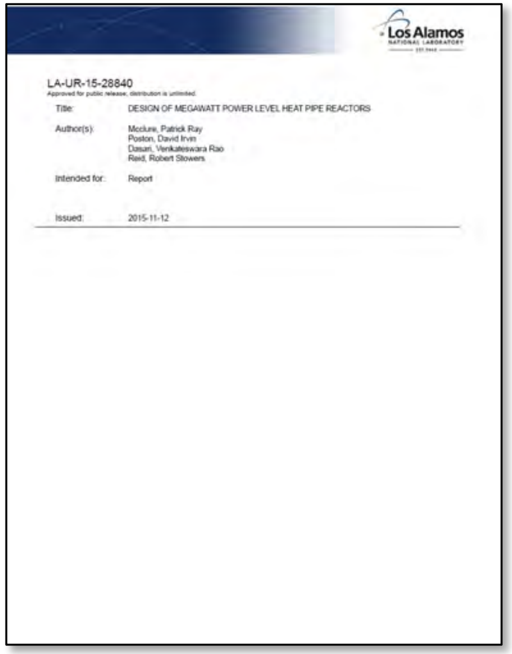


Figure 2-11. KRUSTY Core and Core Assembly

Publicly available designs

LANL Megapower

- Megawatt-size heat pipe reactor
- Described in LA-UR-15-28840


Los Alamos

LA-UR-15-28840

Approved for public release; distribution is unlimited.

Title: DESIGN OF MEGAWATT POWER LEVEL HEAT PIPE REACTORS

Author(s): Mcclure, Patrick Ray; Poston, David Irvin; Dasari, Venkateswara Rao; Reid, Robert Stowers

Intended for: Report

Issued: 2015-11-12


Preliminary Assessment of Two Alternative Core Design Concepts for the Special Purpose Reactor

James W. Sterbentz, James E. Werner, Andrew J. Hummel, John C. Kennedy, Robert C. O'Brien, Axel M. Dion, Richard N. Wright, Krishnan P. Ananth

May 2018

INL Idaho National Laboratory

INL Designs A and B

- Two alternatives to Megapower for improved performance and ease of construction
- Described in INL-EXT-17-43212, Rev 1

LANL Megapower versus INL Design A

LANL Megapower

- UO_2 High-Assay Low-Enriched Uranium (HALEU) fuel
- Fuel region contained between the top and bottom reflector assemblies
- Negative temperature coefficient from Doppler broadening and axial elongation
- Passive removal of decay heat

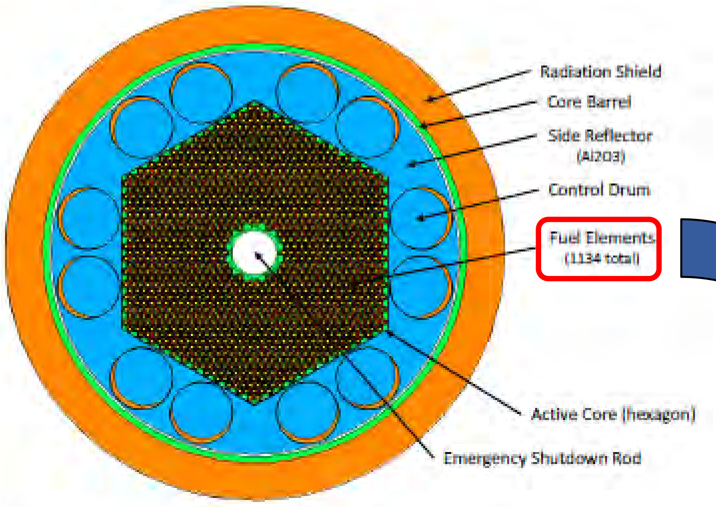
INL Design A

- To address potential issues with manufacturing and defense in depth
  - No stainless-steel monolith (reduces thermal stress, intended to simplify construction)
  - The fuel is encased in stainless steel cladding
  - Heat pipes fabricated separately and inserted into central hole in fuel element
- **Used for SCALE/MELCOR demonstration project**

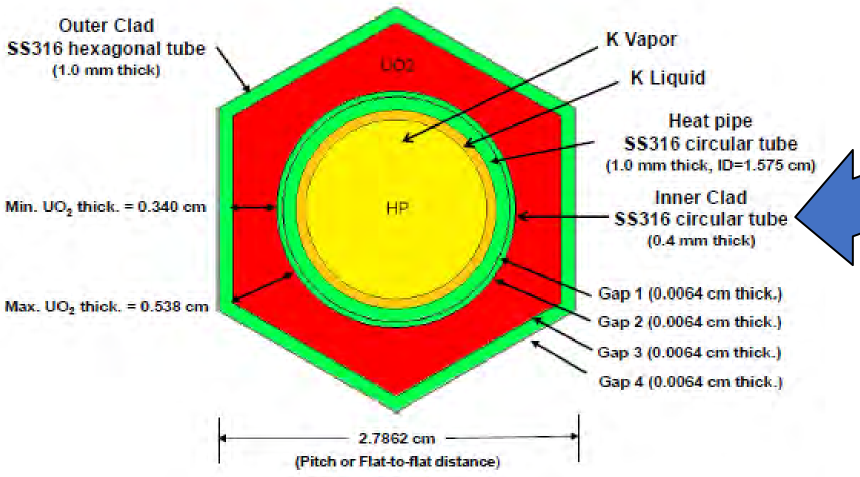
INL Design A (1/2)

Reactor

- 5 MW thermal power
- 1134 fuel elements
  - UO_2 HALEU fuel (5.2 MT)
  - Annular fuel elements with with stainless steel cladding on both sides
  - Outside of fuel element has hexagonal shape
  - HP at the center of each fuel element
- 1134 heat pipes
  - Potassium at 650 to 750 C
  - Vertical orientation for gravity-assisted performance
  - 1.8 cm outside diameter
- 2 emergency control rods of B_4C
- 12 alumina control drums with arcs of B_4C for reactivity control



Cross-sectional view of the Design A core layout.



INL Design A (2/2)

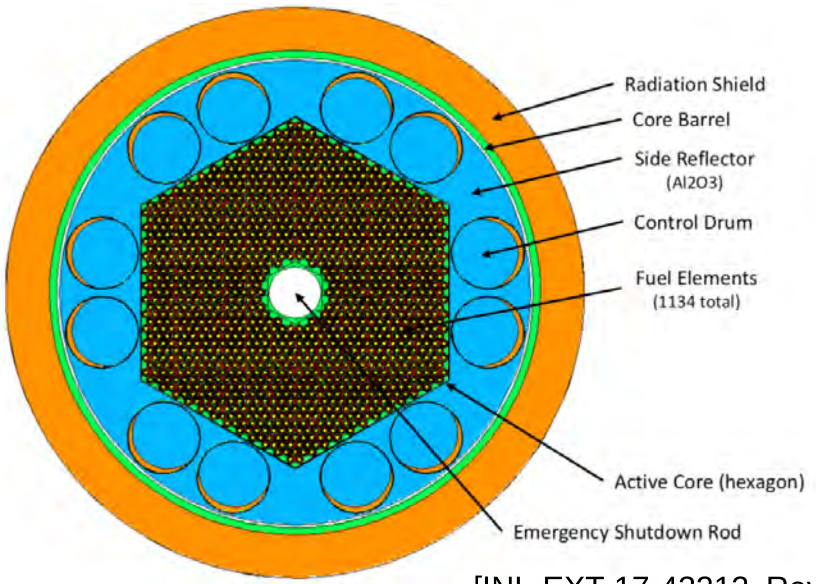
Reactor

- 3 neutron reflectors (top, side, bottom) around the core
  - Top/bottom reflectors are stainless steel + beryllium oxide (BeO)
  - Side reflector is alumina (Al_2O_3)
- Radiation shield surrounds the core
  - 5.08 cm stainless steel core barrel
  - 15.24 cm B_4C neutron shield

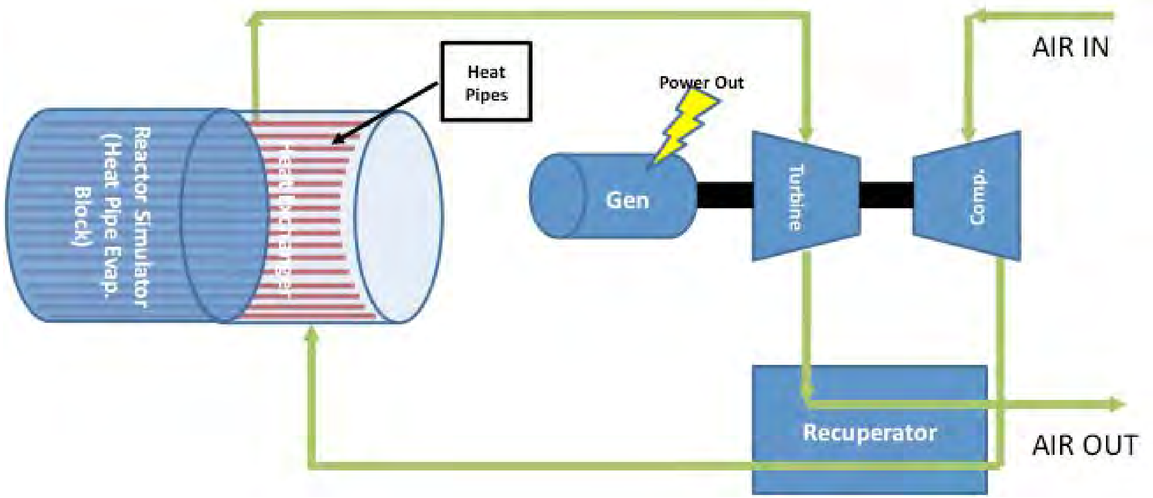
  Reactor

Secondary system

- Open-air Brayton cycle
  - Operates at 1.1 MPa
  - 1.47 MW electrical power output (29% efficiency)



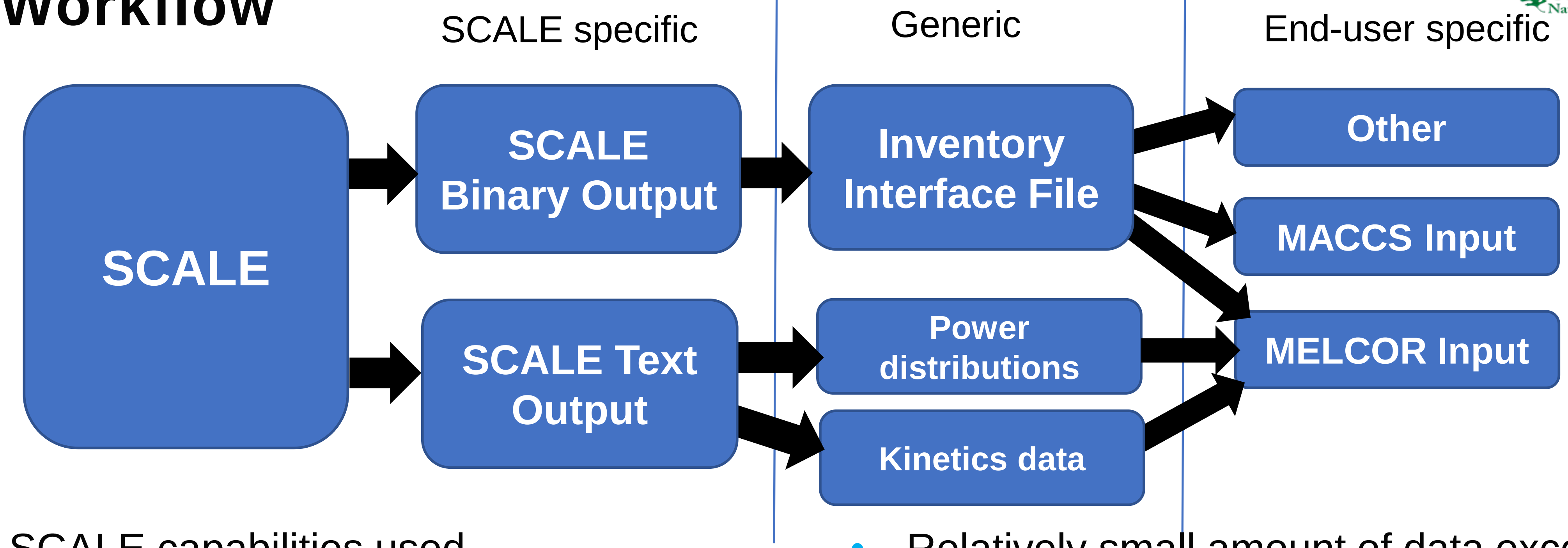
[INL-EXT-17-43212, Rev 1]



[INL/CON-17-41817]

Heat Pipe Reactor Fission Product Inventory/Decay Heat Methods and Results

Workflow

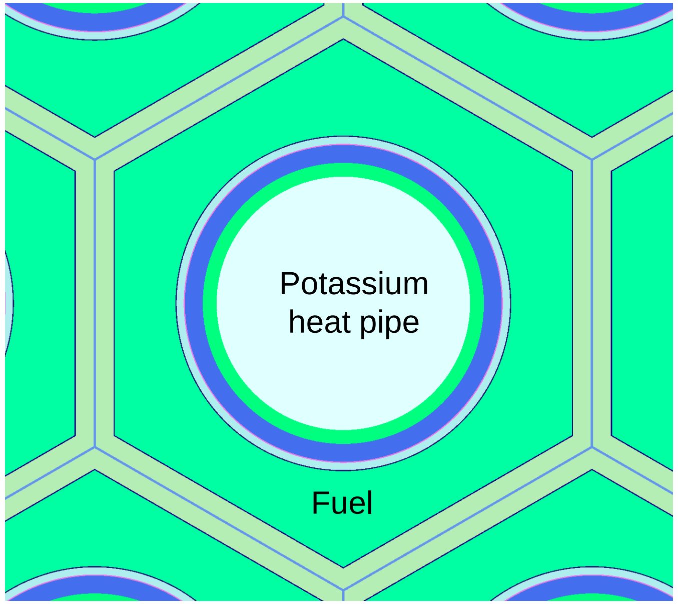


- SCALE capabilities used
  - KENO or Shift 3D Monte Carlo transport
  - ENDF/B-VII.1 continuous energy physics
  - ORIGEN for depletion
  - Sequences
    - CSAS for reactivity (e.g. rod worth)
    - TRITON for reactor physics & depletion

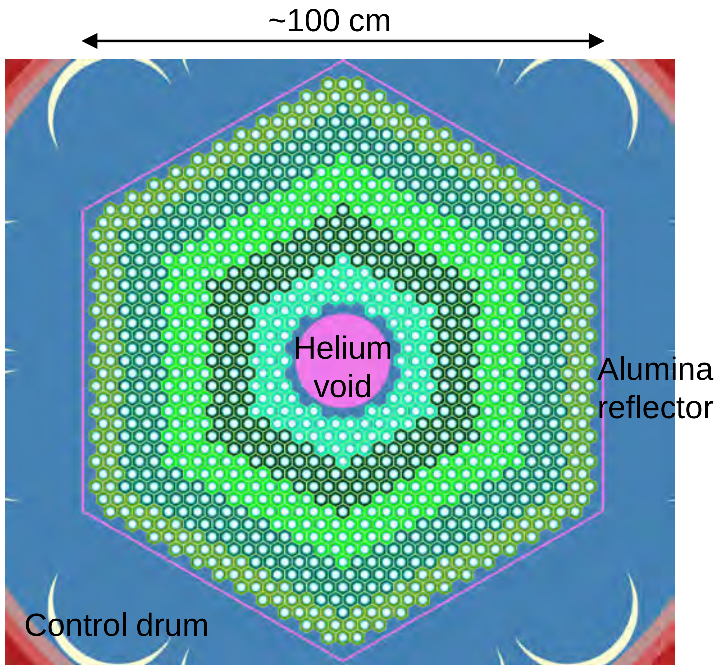
- Relatively small amount of data except for nuclide inventory
  - new interface file developed for inventory using standard JSON format
  - easily read in python and post-processed into MELCOR or MACCS input
  - contains nuclear data such as decay Q-value for traceability when performing UQ studies

INL Design A Neutronics Summary

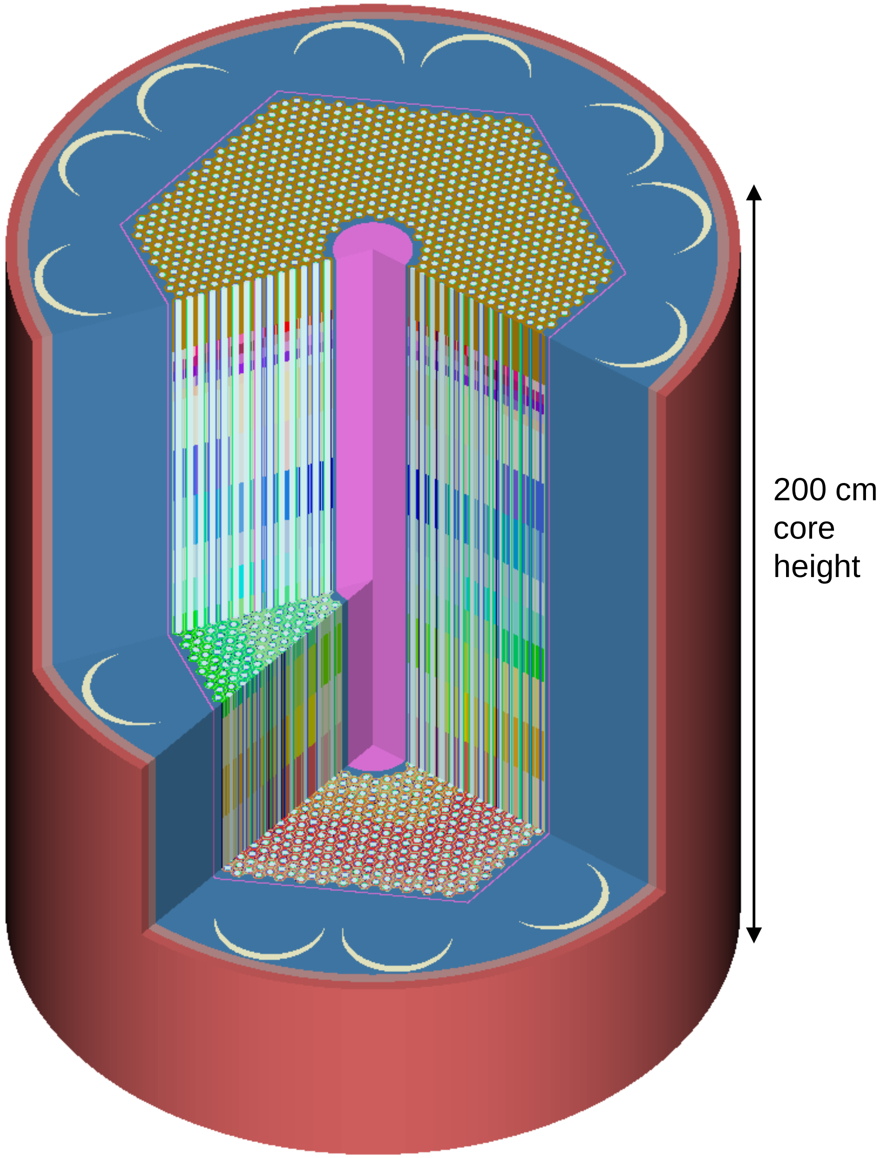
- 5 MWt rated power with 5-year operating lifetime
- UO_2 fuel with 19.75% ^{235}U enrichment
- 4.57 MTU initial core loading
- 1.0951 MW/MTU specific power
- 2.0 GWD/MTU discharge burnup
- 1,134 heat pipe/fuel element units
- Discretized with 20 axial and 5 radial fuel zones



Fuel Element Lattice



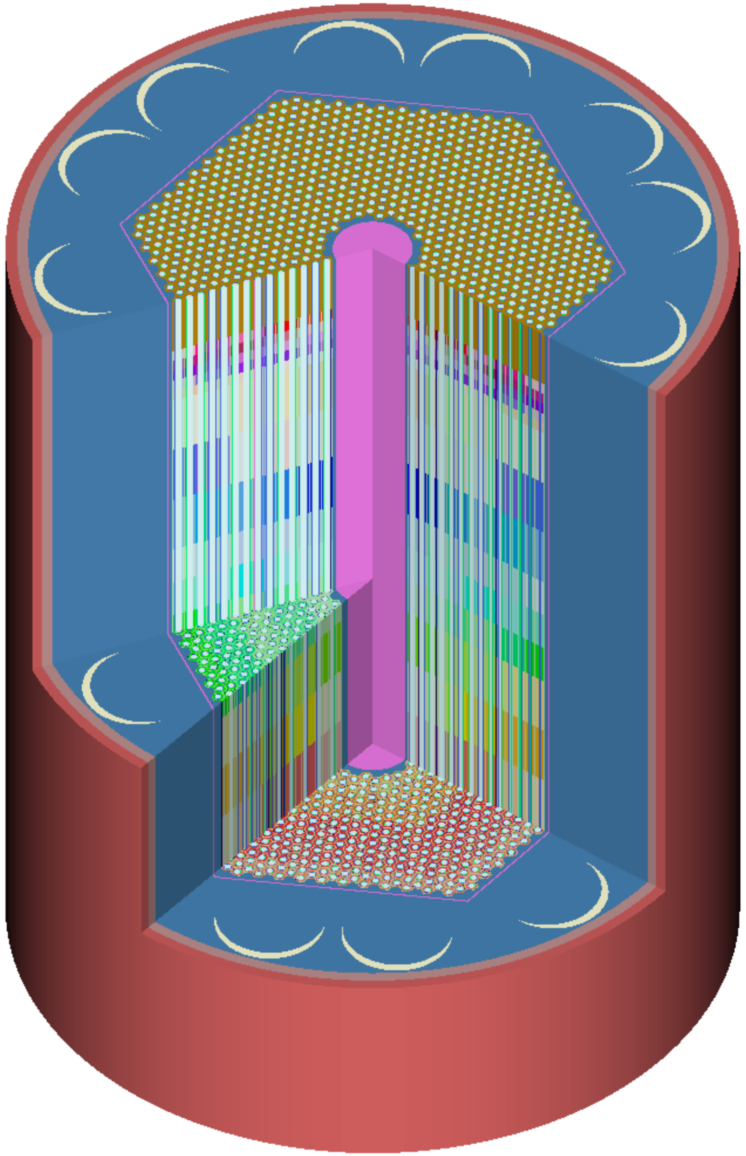
Cross Section at Midline



3D Core Model

HPR-Related SCALE Updates

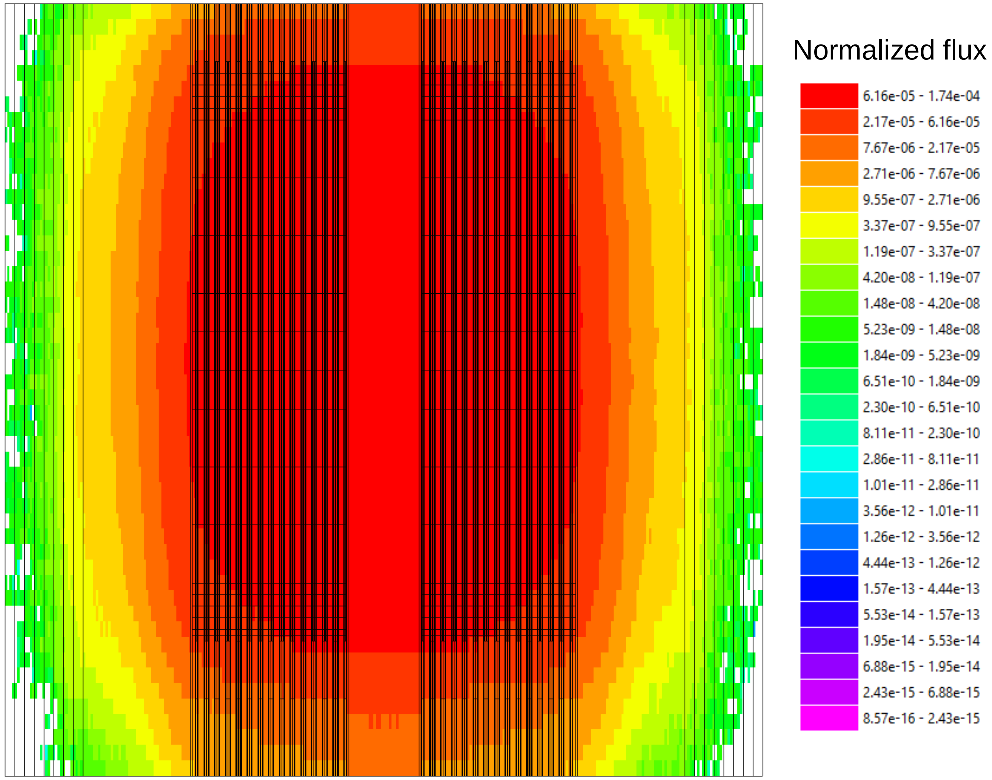
- New fast-spectrum nuclear data library
  - New 302-group structure was developed based on group structures optimized for fast systems (sodium-cooled fast reactors)
  - Enables fast depletion
    - ~6.6 hours → ~1.12 hours using KENO
- Added 3D data visualization
  - Input geometry
  - Mesh data overlay (flux, fission source)
- Probability table update for unresolved resonance region for fast systems
  - ~400 pcm error for fast-reactor systems with Pu
- Shift integration
  - Full-core continuous-energy (CE) depletion is tractable for HPRs and TRITON-Shift scales to 10,000's of cores for faster turnaround (TRITON-KENO only scales to <100's of cores)
- Developed MADRE test suite for advanced reactors
  - Finds equivalent multigroup (MG) vs. CE performance
  - ENDF/B-VIII.0 ~400 pcm less than ENDF/B-VII.1



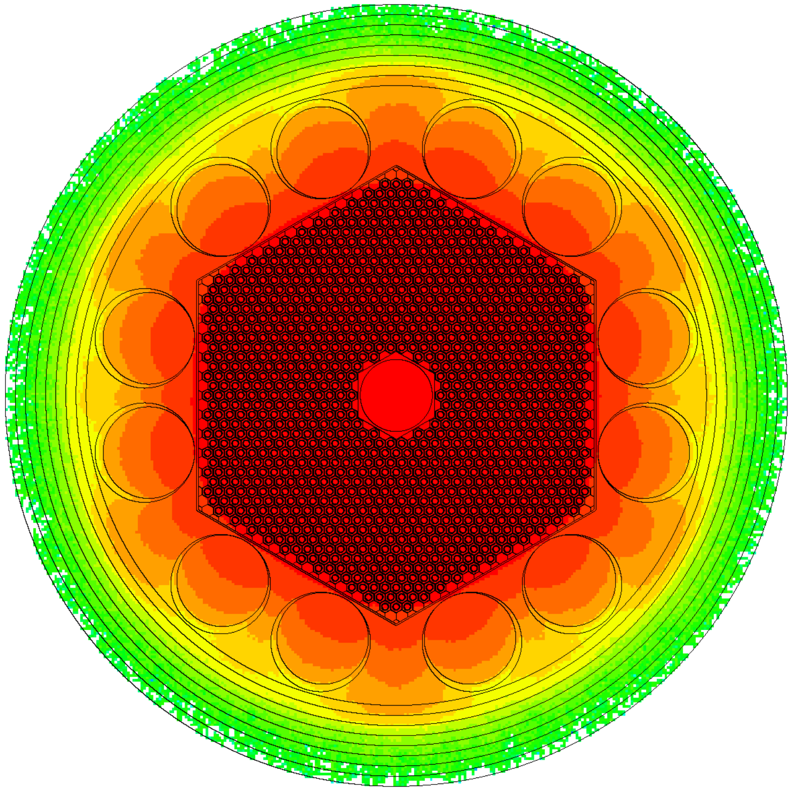
3D SCALE model of INL Design A

HPR-Related SCALE Updates

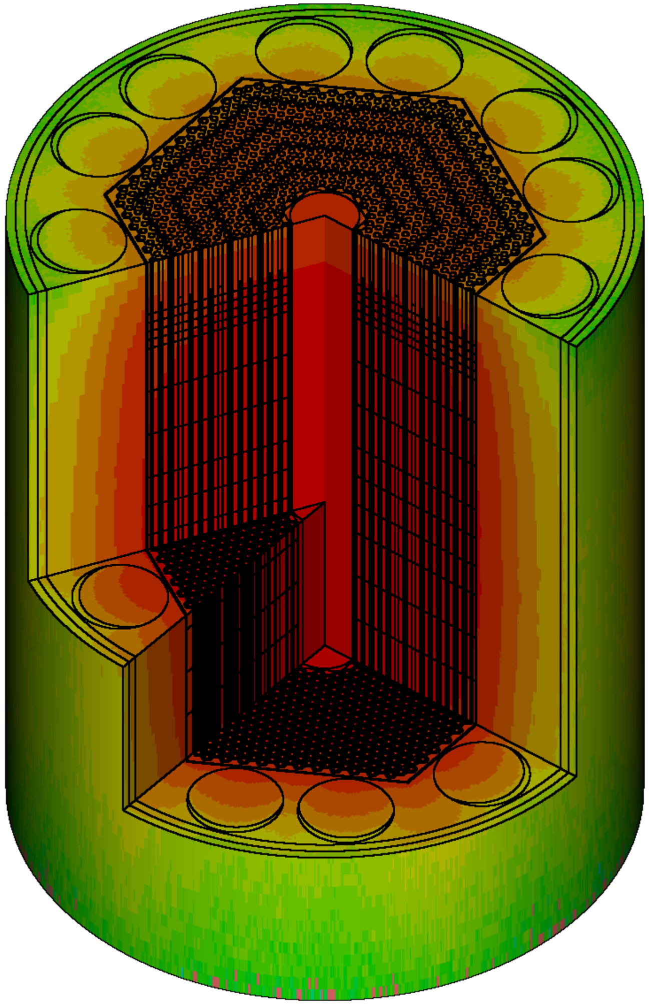
- Example of 3D beginning-of-life (BOL) flux map overlay



Front (X-Z)



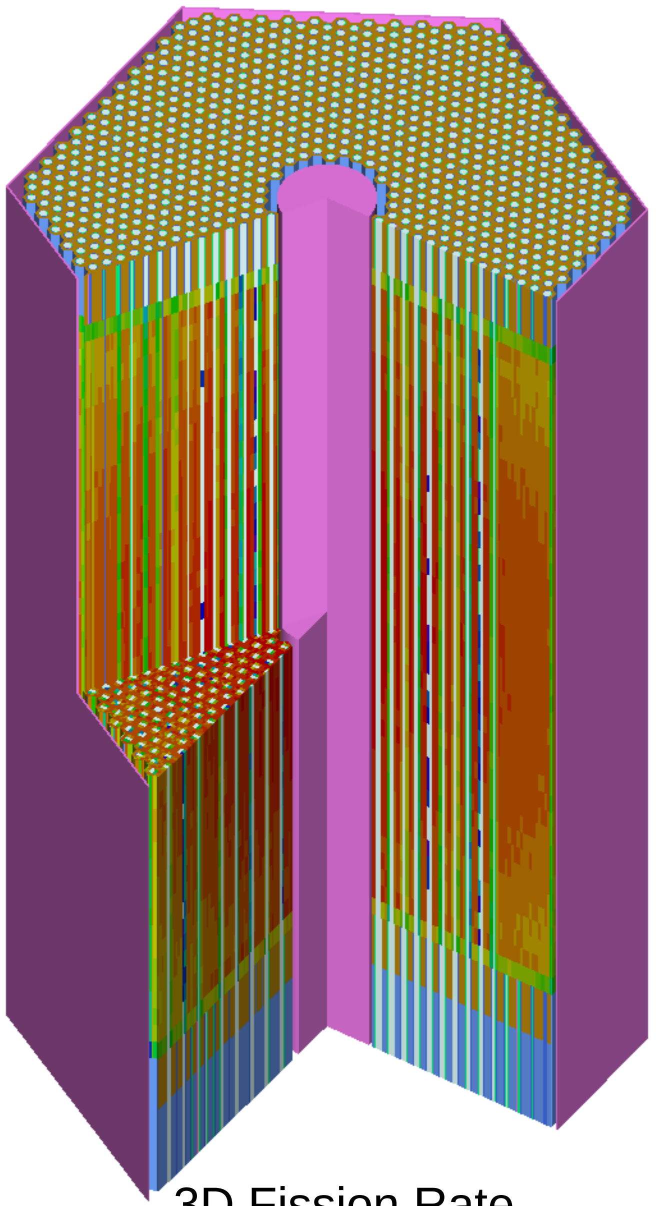
Top (X-Y)



3D

Modeling Assumptions

- Full-core 3D Monte Carlo with continuous energy physics
- System state defined in INL report
  - Temperature
    - Fuel
      - iteration 1: uniform 1,000K
      - iteration 2: informed by MELCOR temperature profile
    - Working fluids 950K
    - Reflector 950K
  - Geometry
    - Annular fuel
    - Thermal expansion of
      - fuel stack (UO_2)
      - radial reflector (Alumina)
      - fuel cladding (stainless steel [SS])



3D Fission Rate

Verification and Validation

- Verification
  - Compared to INL A reference design description
    - Axial power shape
    - Control drum worth
  - Multigroup (faster) vs. continuous energy physics (more accurate) comparison shows an average **~150** pcm higher reactivity

Years	0.00	0.42	0.84	1.25	1.67	2.08	2.50	2.92	3.33	3.75	4.17	4.58	5.00
MG-CE Diff (pcm)	141	60	62	260	122	145	104	134	198	149	305	112	128

  - ENDF/B-VIII vs. ENDF/B-VII.1 comparison shows an average **~300** pcm lower

Years	0.00	0.01	0.42	0.84	1.25	1.67	2.08	2.50	2.92	3.33	3.75	4.17	4.58	5.00
VIII.0-VII.1 Diff (pcm)	-272	-302	-288	-295	-315	-288	-298	-327	-306	-309	-259	-288	-280	-333

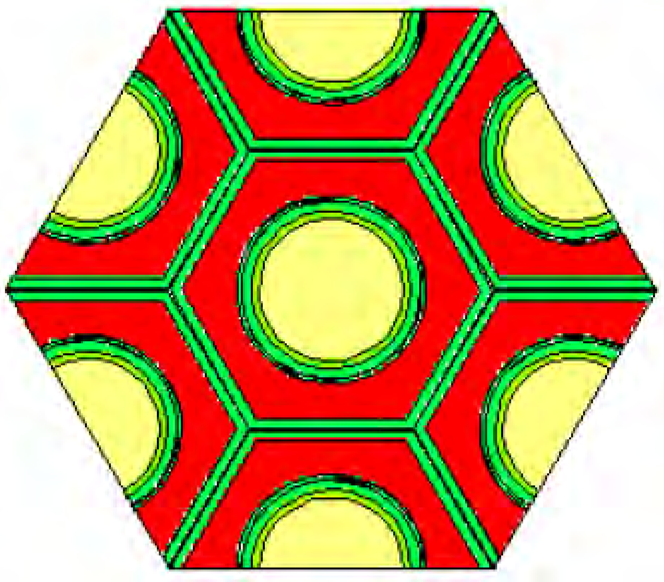
- Validation basis
  - **1% +/- 2% bias in decay heat** based on burst-fission experiments
  - **200 pcm +/- 400 pcm bias in eigenvalue** based on 24 critical experiments with similarity index c_k>0.9 compared to BOL cold zero power

Unit Cell Verification

- INL reported sensitivity study results of k-inf on pitch and clad thickness changes using infinite unit cell models in MCNP
- These models were replicated in SCALE
  - Using identical explicit isotopics with ENDF/B-VII.0 library used in the INL report
  - Using the SCALE standard composition library with ENDF/B-VII.1 library
- SCALE k-inf results differed by roughly 50 pcm with identical models
- Updating library and material definitions added, on average, 50 pcm to the SCALE results

Case	Outer SS Clad (cm)	Pitch (cm)	k_{inf}[1] (MCNP)	k_{inf} (SCALE) ENDF 7.0, Isotopics	k_{inf} (SCALE) ENDF 7.1, Std Comp
1	0.1	2.786	1.25953	1.259937 (40.7)	1.260461 (93.1)
2	0.05	2.786	1.27496	1.275447 (48.7)	1.275798 (83.8)
3	0.05	2.686	1.28830	1.288864 (56.4)	1.289494 (119.4)

1. "Preliminary Assessment of Two Alternative Core Design Concepts for the Special Purpose Reactor", NL/EXT-17-43212, May 2018

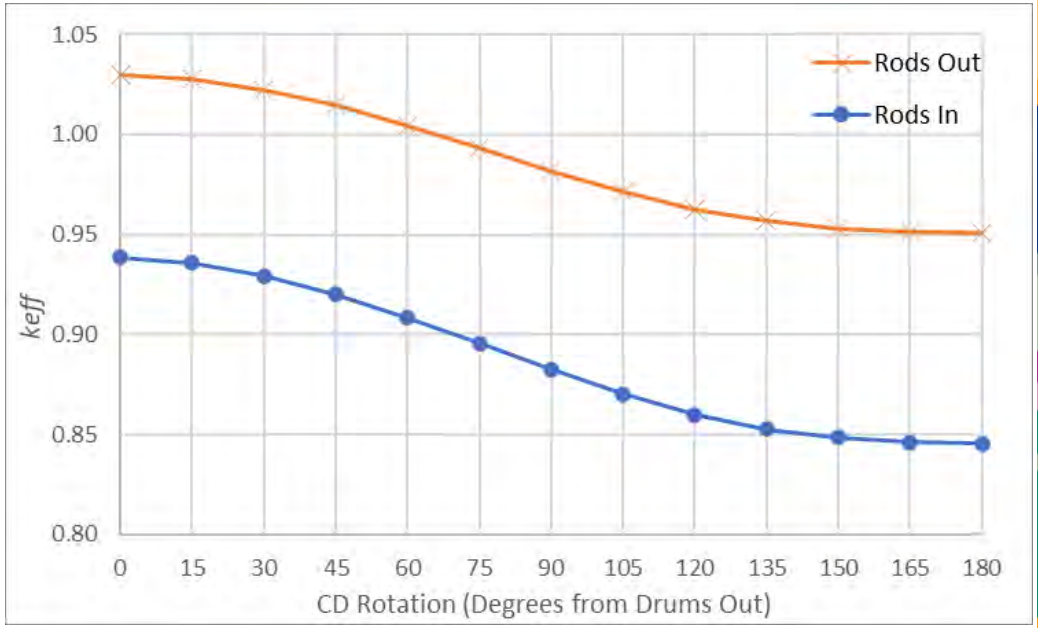


Design A unit cell with reflective boundary conditions[1]

Full Core Verification

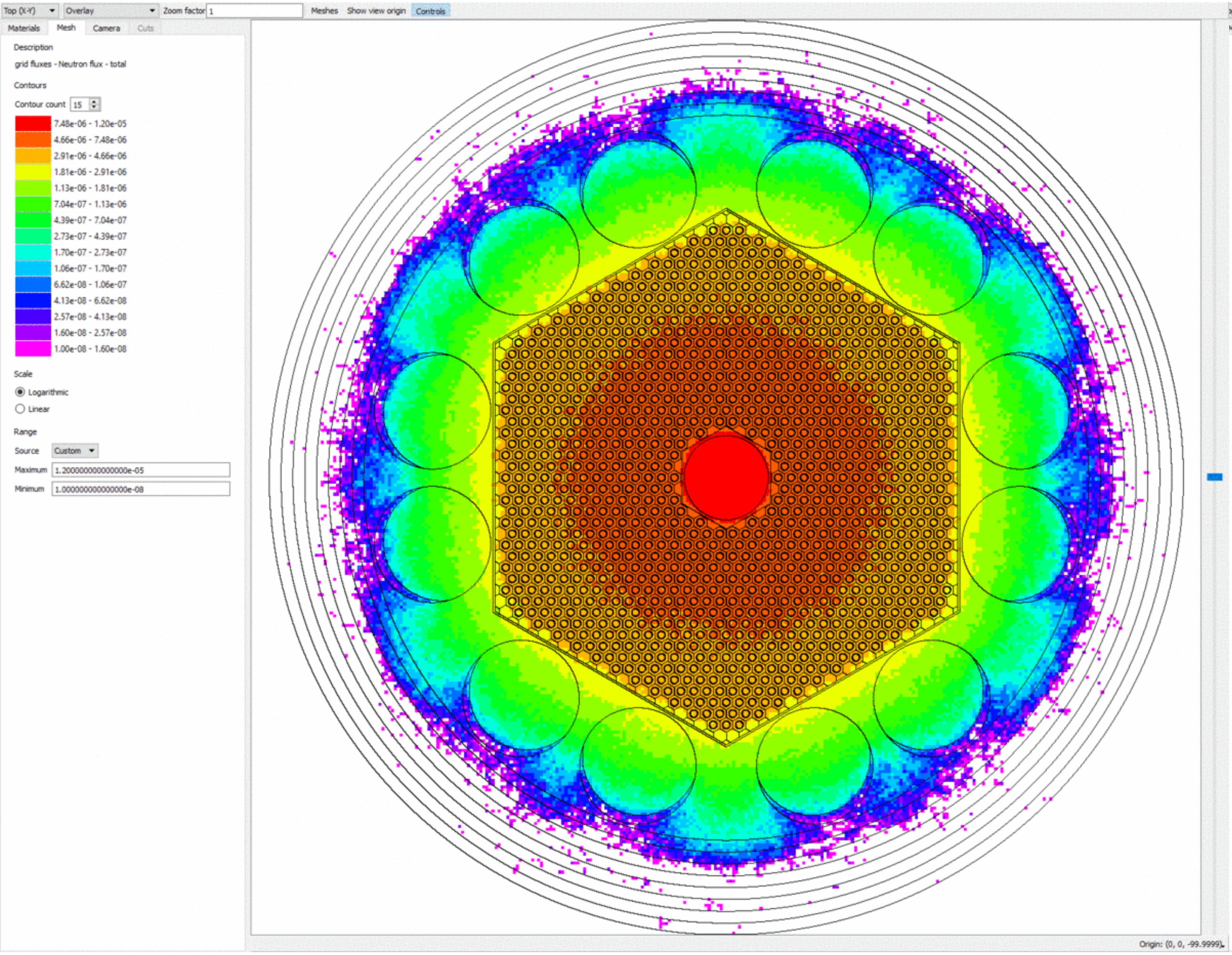
- Reactivity control devices were tested in different configurations
  - All Poisons Out – Both shutdown rods were withdrawn, and control drums (CDs) were turned away
  - All Poisons In – Both shutdown rods were inserted, and CDs were turned in
- Comparisons were done using both the explicit isotopics with ENDF/B-VII.0 library and standard composition library with ENDF/B-VII.1
- Reactivity worth calculations were performed and compared to reference results
  - Identical models agree well with < 200 pcm k-eff differences and < 3.5% reactivity worth differences

Control Condition/Parameter	Design A MCNP	Design A SCALE ENDF 7.0, Isotopics	Design A SCALE ENDF 7.1, Std Comp
All Poisons Out	1.02825	1.029816 (156.6)	1.02989 (164.0)
All Poisons In	0.84594	0.846039 (9.9)	0.84526 (-68.5)
Control Drums In	0.95042	0.95067 (25.0)	0.950304 (-11.6)
Annular Shutdown Rod In	0.94555	0.947445 (189.5)	0.94725 (170.0)
Solid Shutdown Rod In	0.95933	0.960734 (140.4)	0.960660 (133.0)
	β=0.007	β=0.0072 (20)	β=0.0072 (20)
BOL Excess Reactivity ($)	3.925	4.021 (2.5%)	4.025 (2.6%)
Total Drum Worth ($)	11.377	11.228 (-1.3%)	11.278 (-0.9%)
Individual Drum Worth ($)	0.970	0.985 (1.5%)	0.990 (2.1%)
Annular Shutdown Rod Worth ($)	12.151	11.725 (-3.5%)	11.749 (-3.3%)
Solid Shutdown Rod Worth ($)	9.981	9.698 (-2.8%)	9.705 (-2.8%)

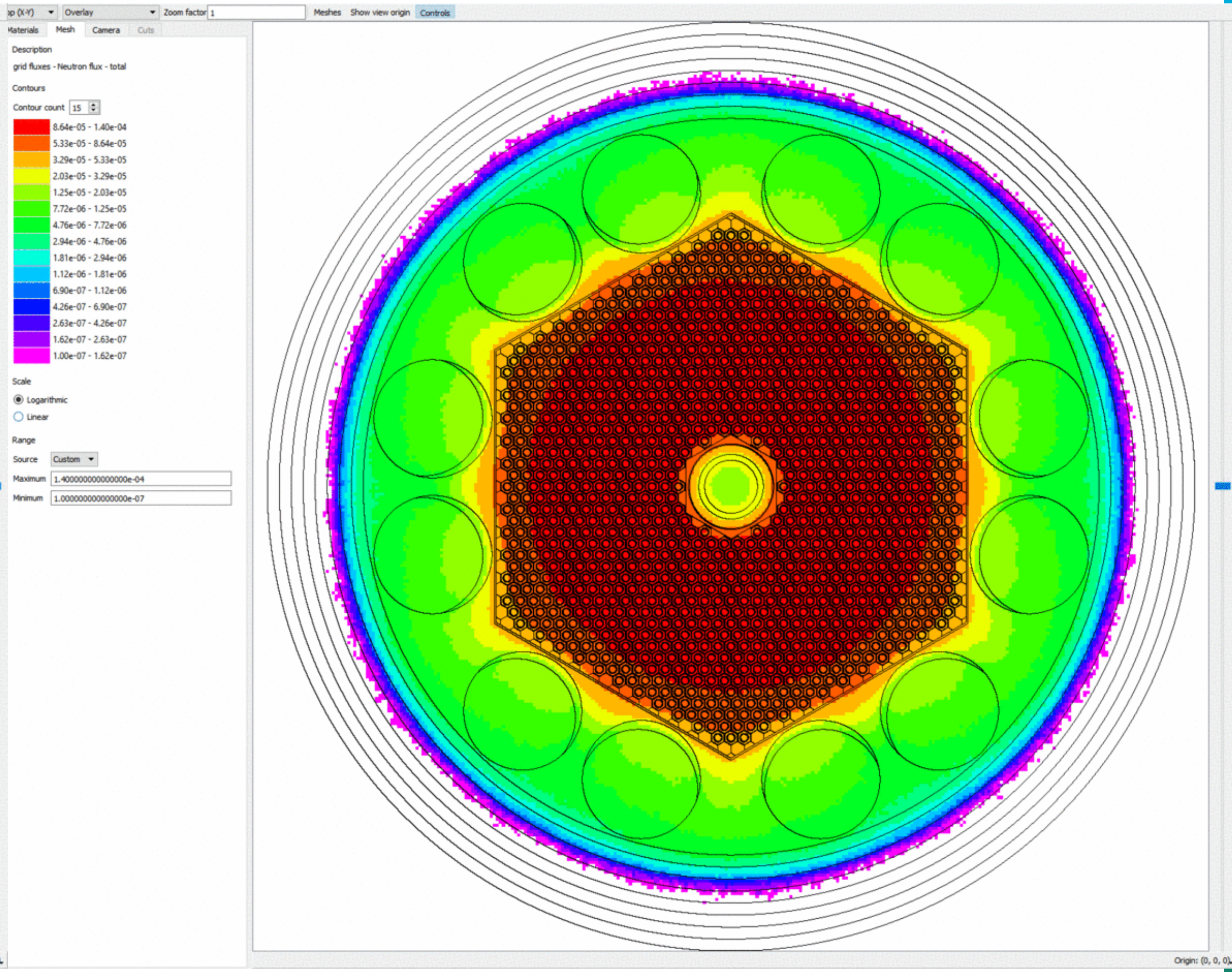


Effect of control drum rotation on eigenvalue

Control Drum Rotation Flux Animations

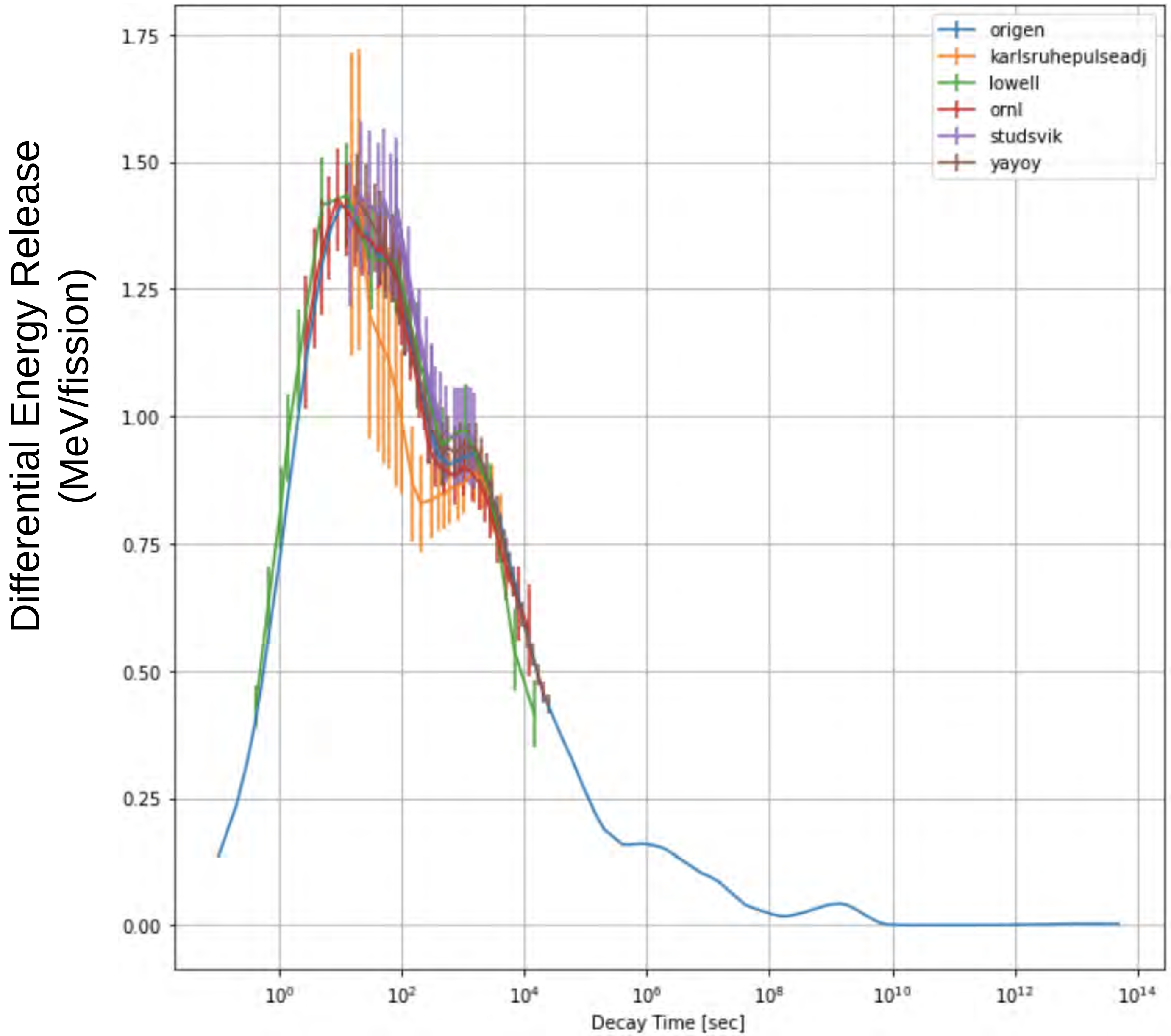


Shutdown rods out



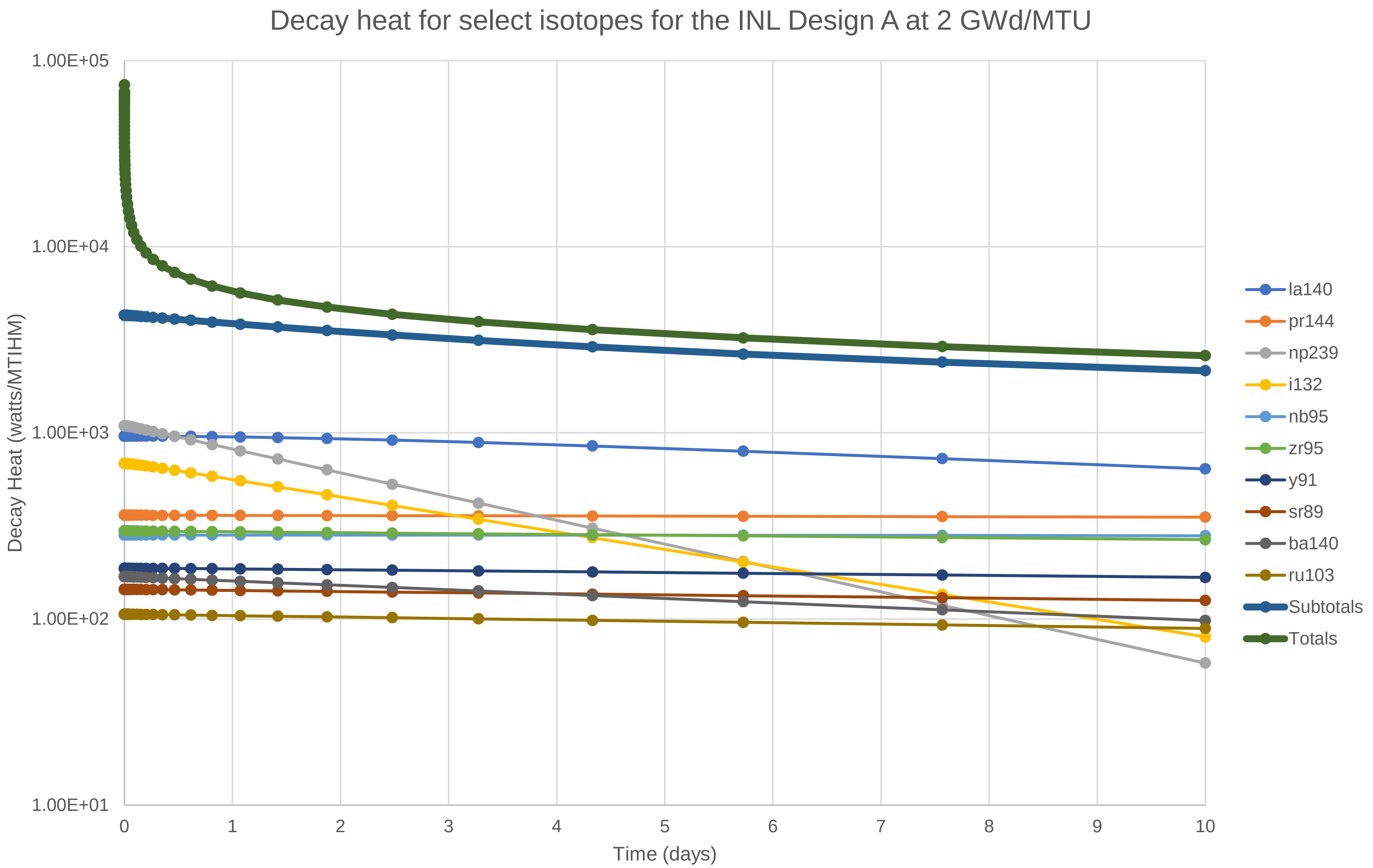
Shutdown rods in

Validation Basis: Short-Term Decay Heat



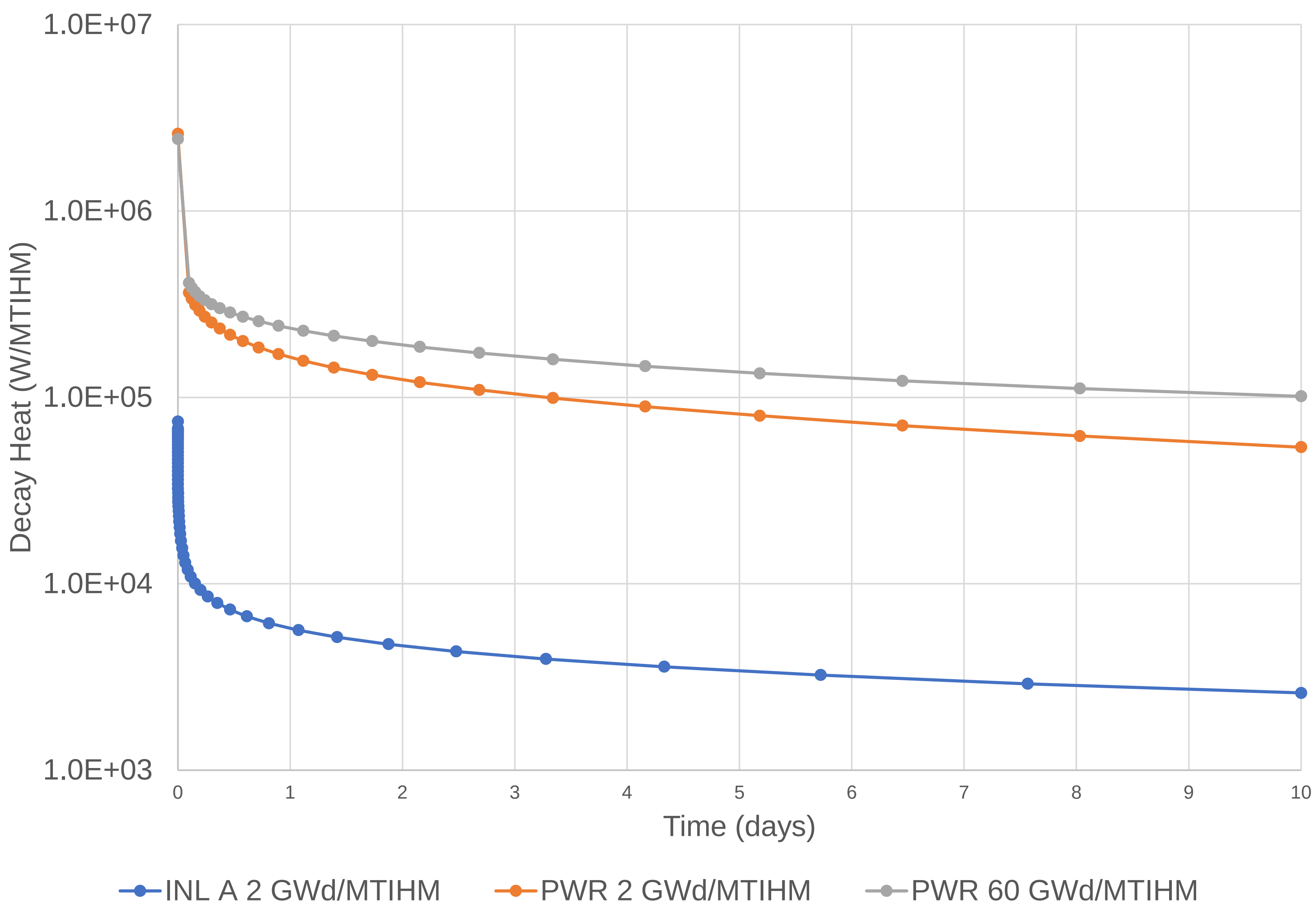
- Fissioning nuclides in INL A
  - 90% from ^{235}U
  - 10% from ^{238}U
  - Negligible from Pu
- Cumulative energy release following shutdown
  - ~90% by 0.3 days
  - ~92% by 1 day
  - ~96% by 10 days
- “Burst fission” experiments measure energy release over time (t<1 day) from a single fission of ^{235}U
  - Most accurate measurements in the set have 1-sigma uncertainty in the 2–3% range
  - ORIGEN simulation is within 2-sigma uncertainty bounds shown in figure for almost all measurements
- **Based on burst-fission data analyzed so far, 1% +/- 2% bias in instantaneous decay heat recommended for SCALE modeling of INL A**

Decay Heat after Core Shutdown



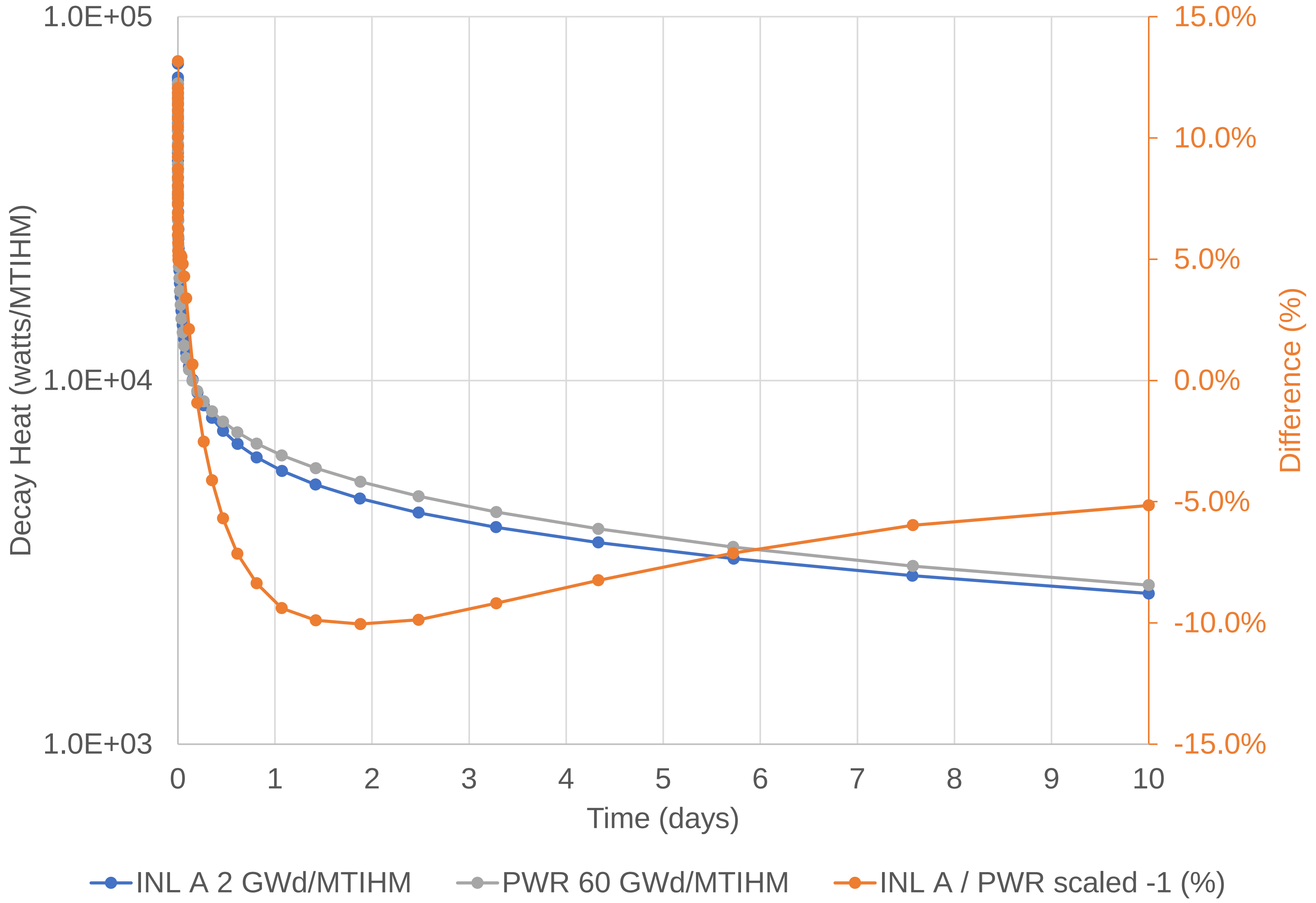
- Top 10 decay heat producing isotopes in the first 10 days following shutdown
- Subtotal shows sum of top 10

Why Is Decay Heat So Much Lower than a Pressurized Water Reactor (PWR)?



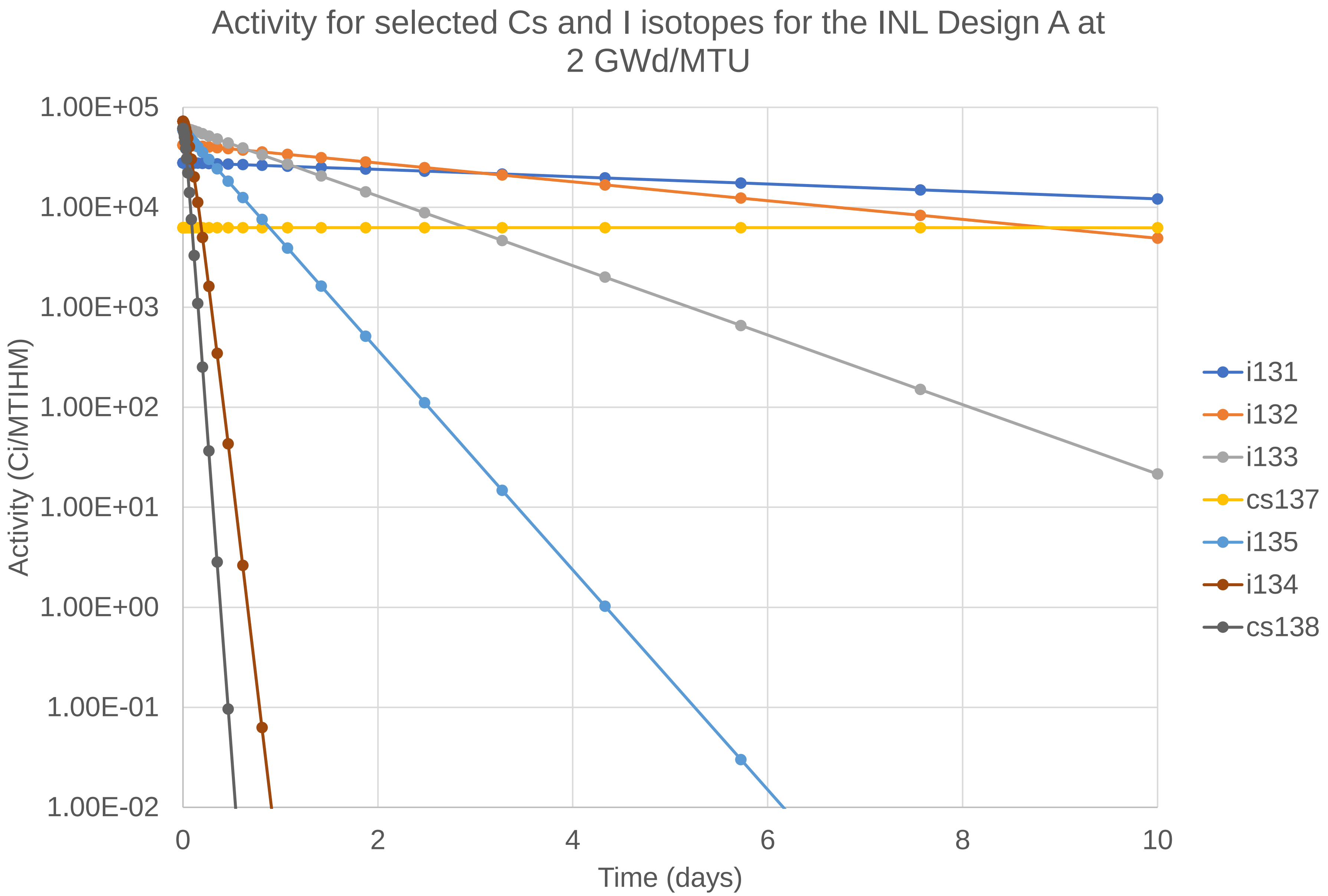
- INL A has 2.7% specific power of PWR
- Comparing INL A fuel vs. PWR fuel **per MTIHM**
- PWR at 2 GWd/MTU
  - INL A has 2.9% of PWR decay heat at t=0
  - INL A has 4.8% of PWR decay heat at t=10 days
- PWR at 60 GWd/MTU
  - INL A has 3.1% of PWR decay heat at t=0
  - INL A has 2.6% of PWR decay heat at t=10 days
- Does this mean decay heat can be scaled with specific power?

Scaling HPR Decay Heat Curve from a PWR



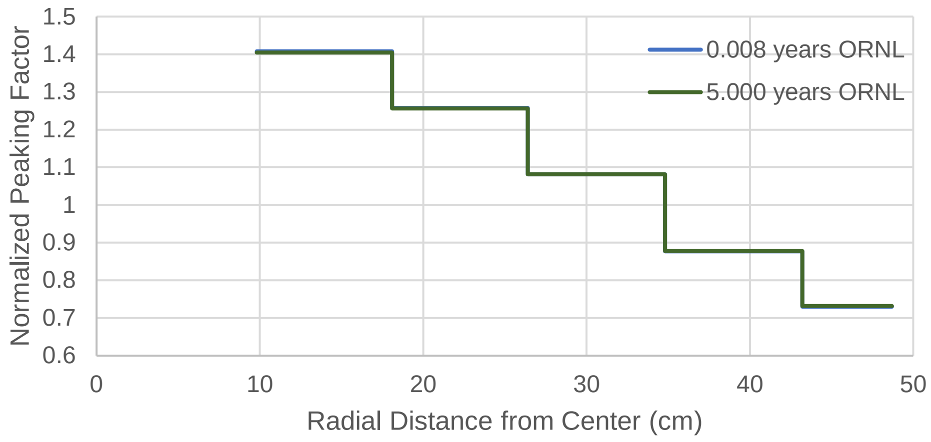
- Comparing INL Design A with PWR decay heat curve scaled down
  - INL A differs by
    - 13.2% at t=0
    - -10.0% at t=1.88
    - -5.2% at t=10
- Decay heat does not scale using specific power

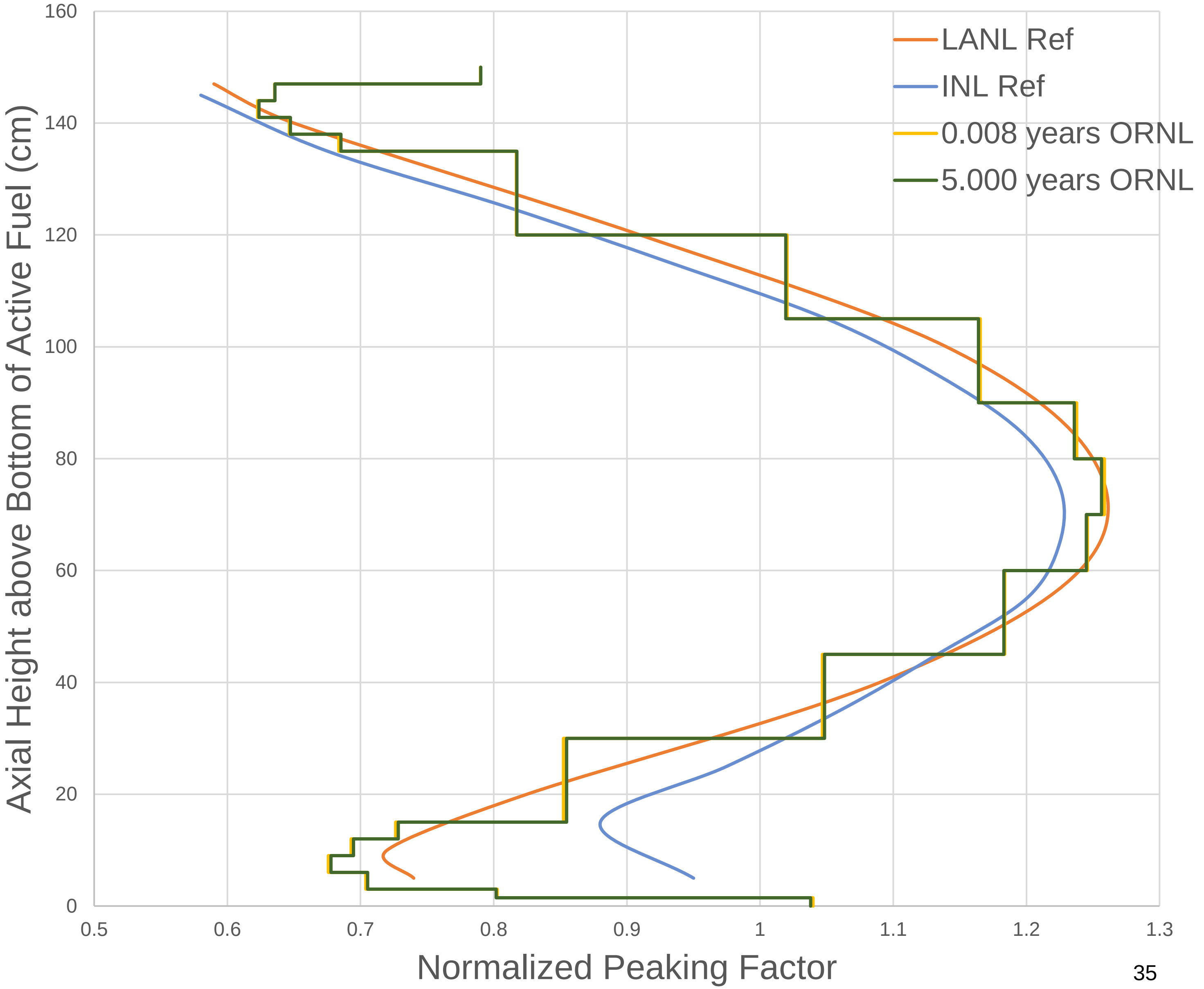
Activity after Core Shutdown



Power Distribution

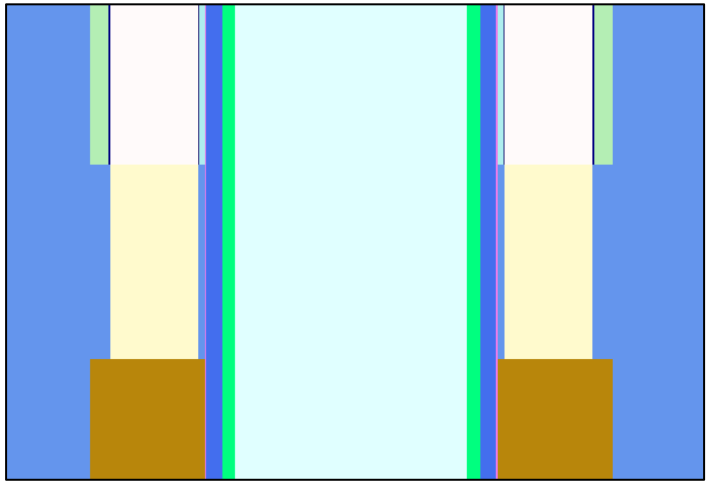
- Axial-normalized power peaking factors agree well with distribution from LANL and INL documents
  - INL reference gives hottest pin power profile while the LANL and ORNL are core averages
- Peaks at the top and bottom are due to axial reflection
  - Not as much reflection in the top due to heat pipes
  - MCNP models did not use fine enough mesh to capture bottom reflector peak fully
- Axial peaking does not fluctuate over core lifetime due to low burnup



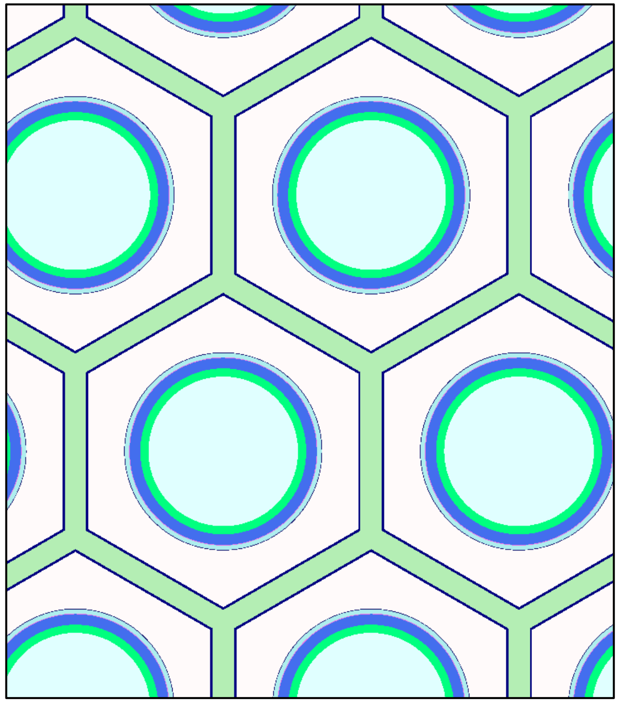


Reactivity Feedback Effects

- Four negative reactivity feedback effects reported
  - Doppler broadening (primary)
  - Fuel axial thermal expansion
  - Alumina reflector radial thermal expansion
  - Outer clad radial thermal expansion
- Modeled all radial effects simultaneously
  - Outer SS clad radial expansion
  - Gap closure and increased pitch
  - Alumina radial expansion
  - Control drum drift



Axial Fuel Expansion



Radial Clad Expansion

Feedback Effect (cents/°C)	INL	ORNL
Doppler	-0.1074	-0.1113
UO_2 Fuel Axial Elongation	-0.0422	-0.0437
Alumina Reflector Radial Thermal Expansion	-0.0225	-0.0284
Outer SS Fuel Clad Thermal Expansion	-0.0323	-
All Radial Expansions (Clad, Reflector, and CDs)	-	-0.0636
Total	-0.2044	-0.2185

Neutronics Summary

- Eigenvalue bias was assessed for BOL cold zero power
  - 200 pcm +/- 400 pcm based on 24 critical experiments with similarity c_k>0.9
- Decay heat bias 1% +/- 2% based on burst-fission measurements
- New 302-group structure was developed
  - Demonstrates a ~150 pcm bias over core lifetime compared to CE
- Axial refinement study shows higher reflector peaks at top and bottom of core compared with reference documents
- Using SCALE gives a more realistic representation of decay power than scaling PWR decay power
  - 13.2% at shutdown
  - -5.2% after 10 days

MELCOR Heat Pipe Reactor Model

MELCOR Heat Pipe Reactor Modeling: 1

When present, HPs replace conventional convective heat transfer between the fuel and coolant channel with the energy transfer from the fuel to the evaporative region of the HP.

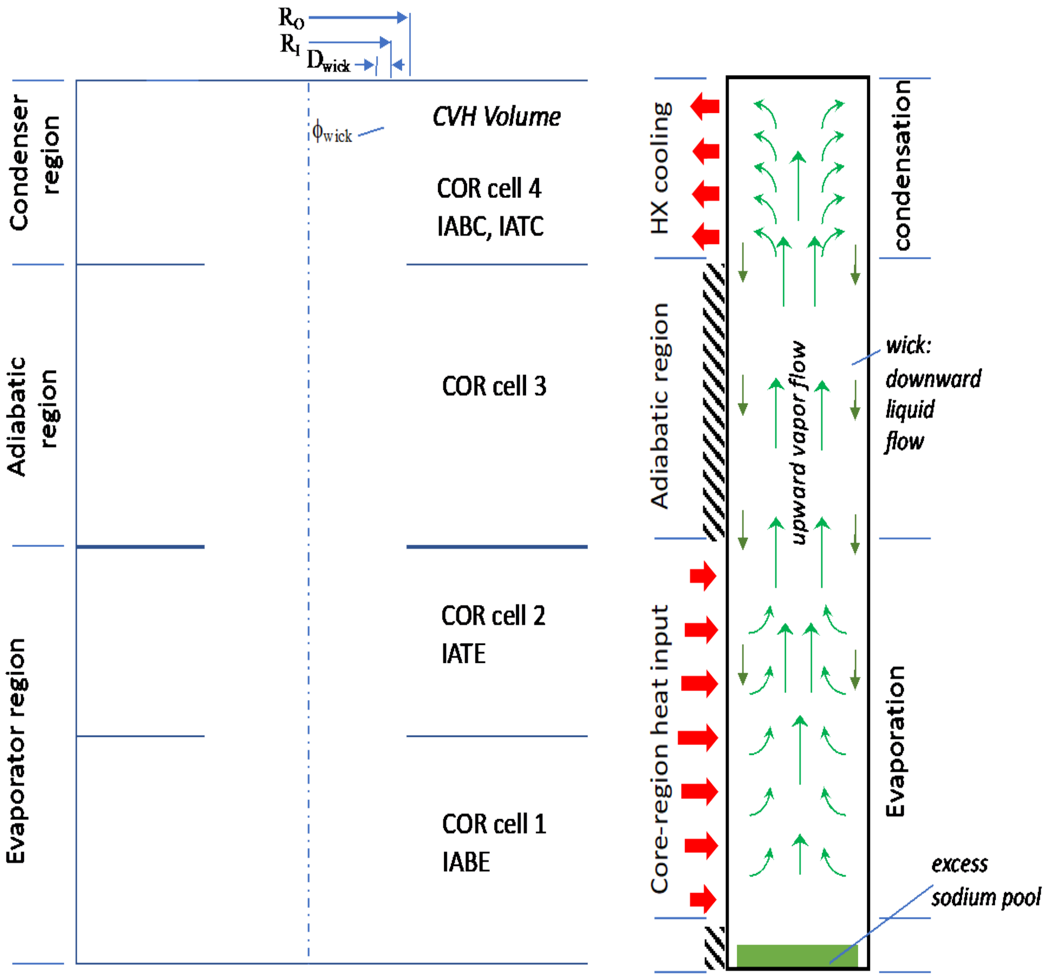
HP models are special components within the COR package.

Heat rejection from the HP model at the condensation interface is transferred to the CVH package.

Basic geometry of a heat pipe is assumed to be a circular cylinder characterized by a relatively small set of geometric values, e.g.:

- R_O outside radius of heat pipe wall (m),
- R_I inside radius of heat pipe wall (m),
- D_{wick} thickness (or depth) of the wick (m), and
- ϕ_{wick} porosity of the wick (-).

Axial lengths of the condenser, adiabatic, and evaporator sections are implicitly defined by the COR package cells that these regions are associated with.



A Generic Heat Pipe

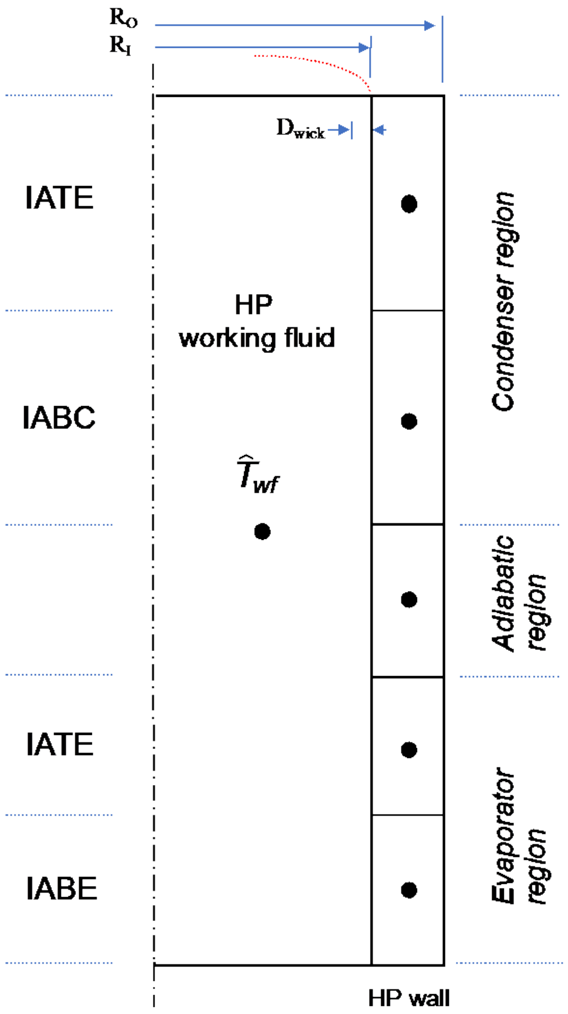
MELCOR Heat Pipe Reactor Modeling: 2

HP modeling approaches within MELCOR reflect the purpose and constraints of the systems-level integrated code that it is.

MELCOR accommodates HP models of different fidelity through a common interface and a specified wall and working fluid region nodalization.

- Model 1: working fluid region modeled as high thermal conductivity material.
- Model 2: thermodynamic equilibrium of working fluid (sodium or potassium EOS). P, T and liquid/vapor fraction evolve in time. Sonic, capillary and boiling limits enforced.
  - Accepts experimental or design-specific performance limit curves
- Flexible implementation allows for multiple HP definitions in the same MELCOR input deck and multiple HP regions

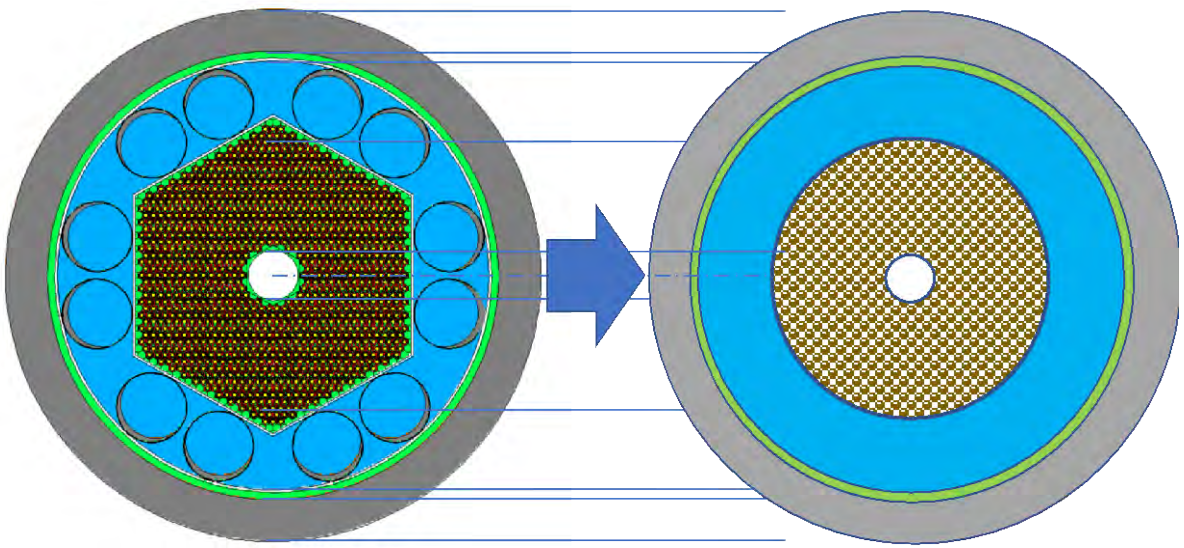
Time-dependent conservation-of-energy equations are solved within the HP component and include boundary conditions linking them with the neighboring fuel (evaporator region) and coolant (condenser region)



Illustrative MELCOR HP component nodalization to define MELCOR variables. Actual nodalization has more nodes.

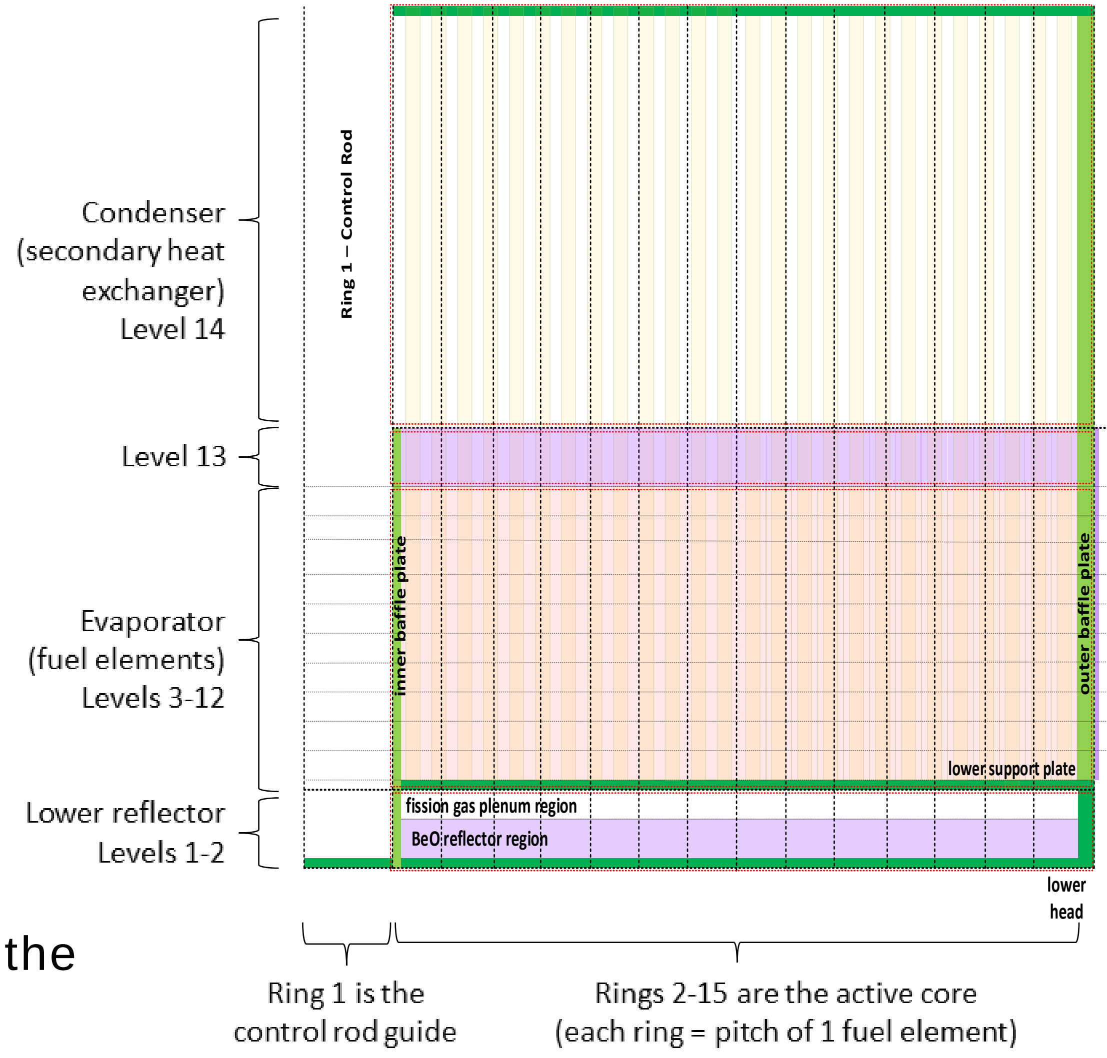
MELCOR Heat Pipe Reactor Modeling: 3

- Core region modeled as a 2-D multi-ring representation



Horizontal “cut” through core region

- Ring-to-ring radiative exchange implemented through the generalized core heat transfer pathway modeling in MELCOR



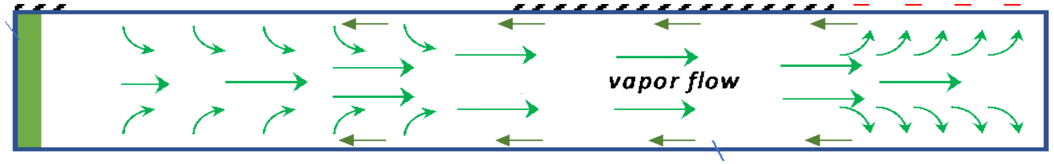
Vertical “cut” through core region

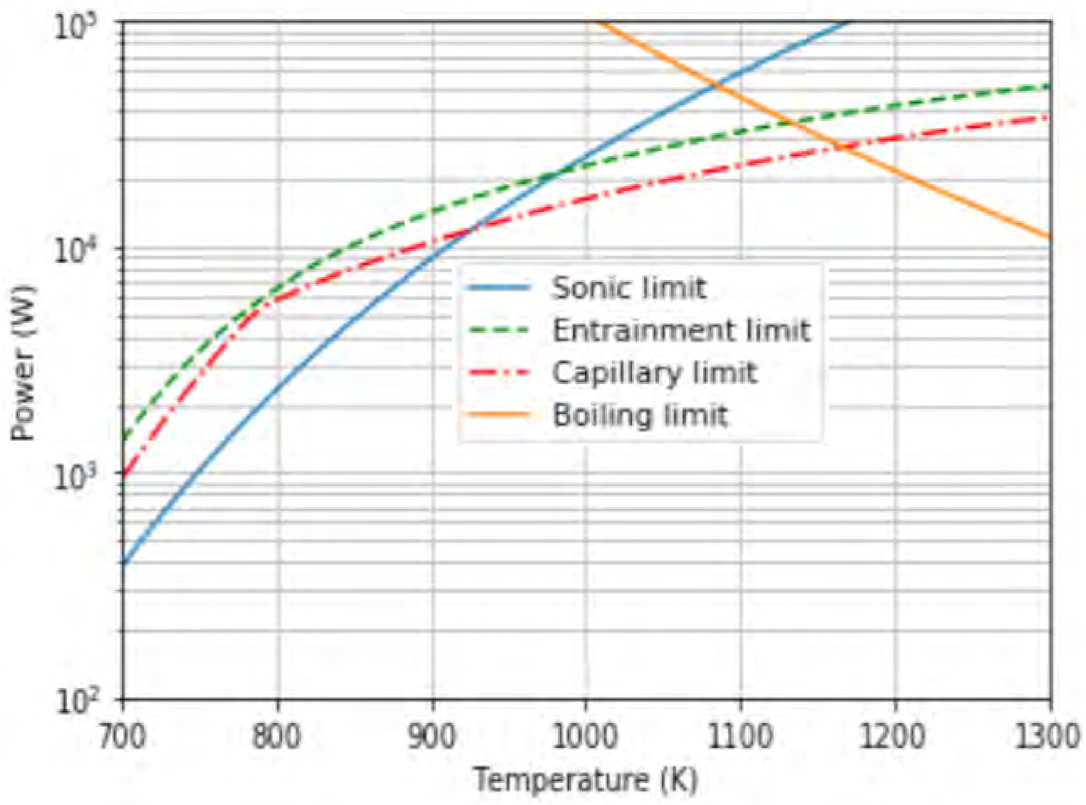
HP limits of operation

Steady-state operational limits modeled in MELCOR

- Sonic limit
  - Choked flow of vapor through the central core
- Capillary flow limit
  - liquid flow rate at maximum capillary pressure difference
- Boiling limits
  - As heat flux increases, both nucleate and film boiling related issues can disrupt heat transfer. Film boiling can lead to a sudden drop in heat transfer efficiency.
- Condenser HX limit
  - The heat exchanger absorbing heat from the condenser may have operation limits of its own.

Each of these limits depend on the HP details (geometry, materials, type of wick, working fluid etc.) and can independently vary in magnitude based on operational conditions.





Estimated HP SS limits for a design considered in a 2018 INL report INL/EXT-17-43212

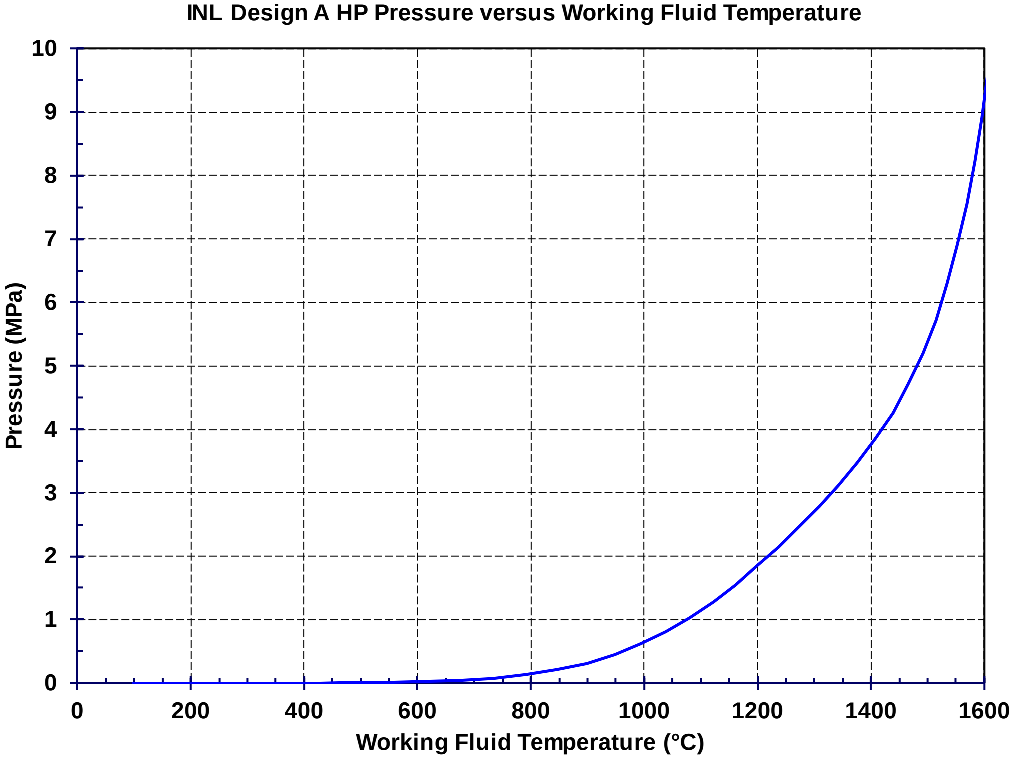
HP Failure Modes Modeled in MELCOR

Several failure modes considered

- HP wall or end-cap failure due to time-at-temperature if the HP is subjected to high operating temperatures and associated pressures, such as might occur in a complete loss of heat sink (e.g., the HX fails)
- Local melt-through of the HP wall due to a sudden influx of heat
- HP wall or micro-imperfections in end-cap welds or HP wall materials after being subjected to time-at-operating temperatures and pressures.

MELCOR HP failure modeling

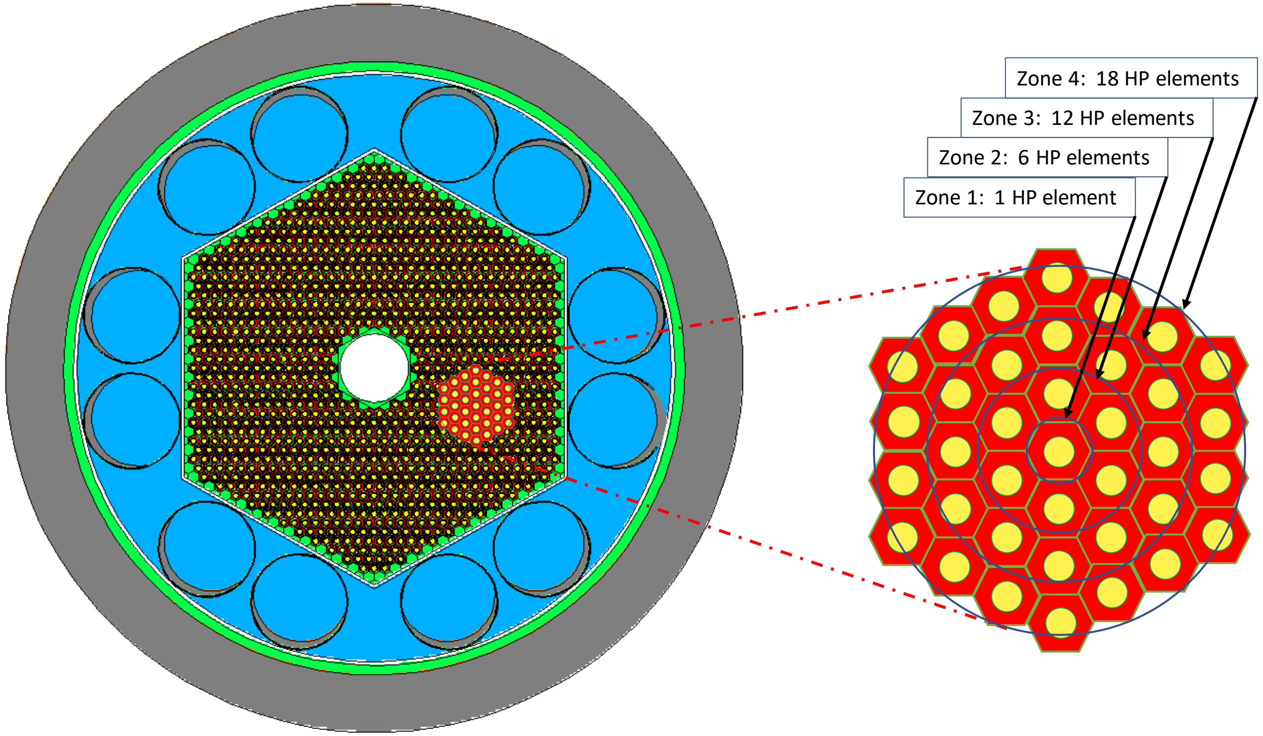
- HP temperature excursion leads to working fluid pressurization and HP wall creep failure
  - o Larson-Miller model used for wall failure
  - o Subsequent response includes HP failure and depressurization
- Alternate user-specified criteria for HP wall failure
  - o HP wall failure can be a specified event (e.g., initiating event) or as an additional failure following a creep rupture failure (i.e., creep failure is predicted before wall melting)
- Optional user features to dynamically control or disable HP evaporator or condenser wall heat transfer and to start the fuel cell radionuclide leakage



MELCOR cascading HP failure modeling

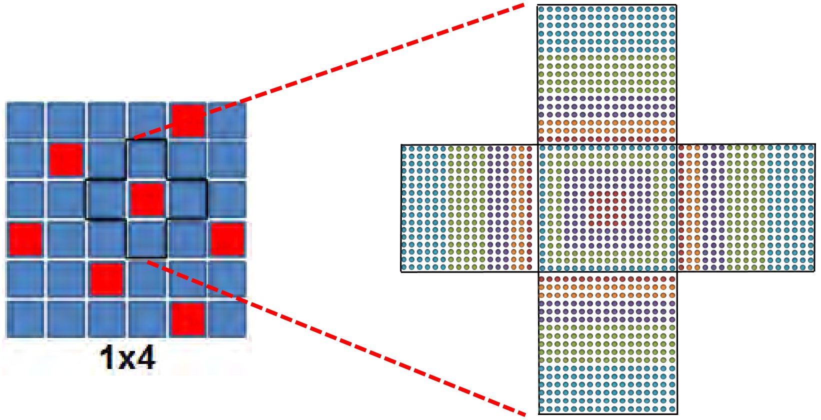
HPR model can be subdivided into an arbitrary number of rings

- Generalized input matrix for fuel element connectivity governs heat flows
- Cascade region shown with 4 zones but could be larger
- User specifies HP failure(s)
- Bulk HP response modeled outside of the cascade region
- Consequences of initial failure(s) on adjacent ring responses



Zone modeling approach used in SFP applications for cascading fuel assembly ignition [NUREG/CR-7216]

Multiple fuel rod components in the center assembly and four peripheral assemblies

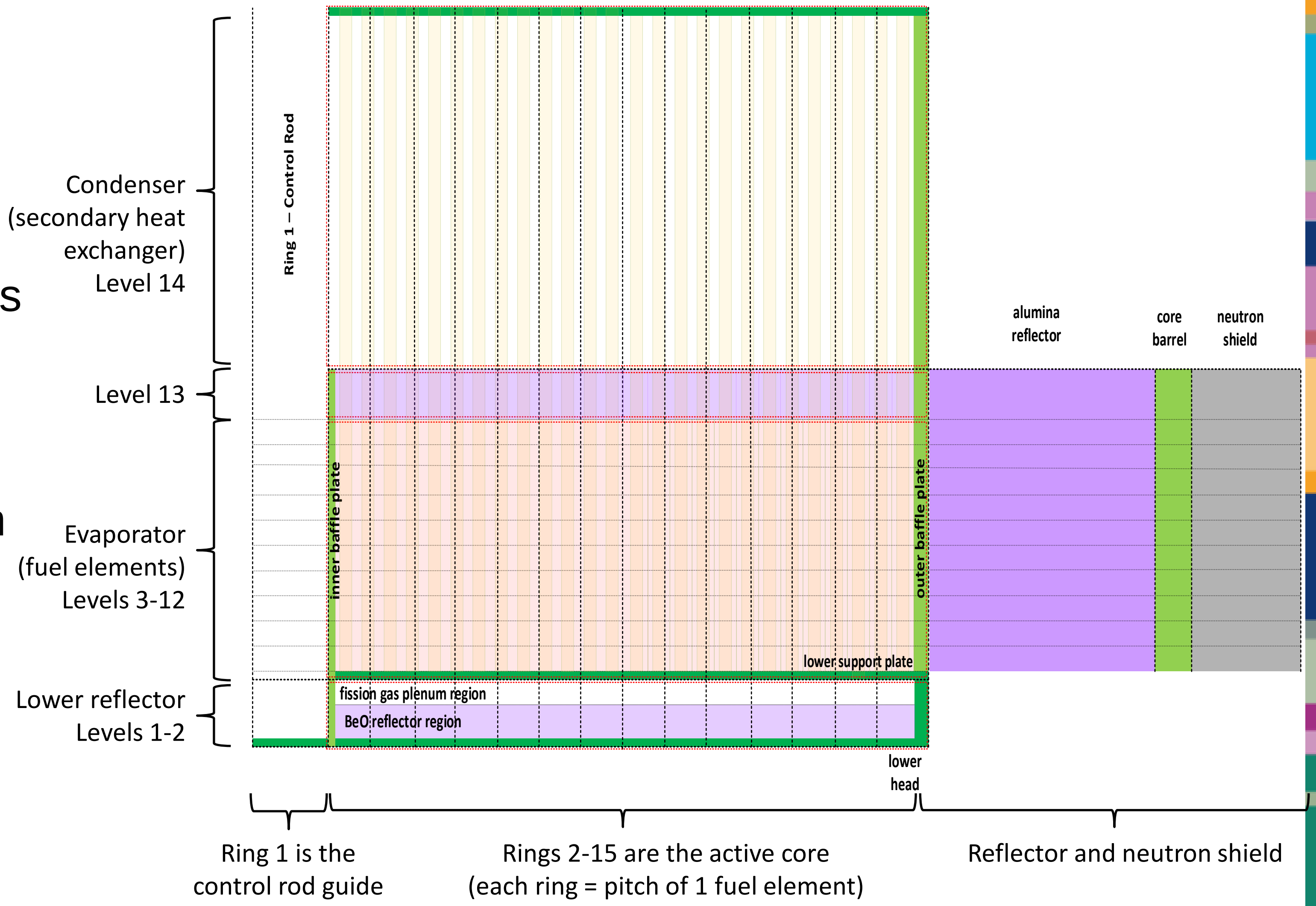


Heat Pipe Reactor Plant Model and Source Term Analysis

MELCOR model of INL Design A – Reactor

Reactor modeling

- 2-D reactor nodalization
  - 14 axial levels
  - 15 radial rings
- 14 concentric rings of heat pipes (width of ~1 fuel assembly)
- Center ring models the emergency control rod guides
- Top and bottom reflectors are in axial levels 1 and 13
- Heat pipes transfer heat to the secondary Brayton air cycle in axial level 14
- Core region is surrounded by stainless steel shroud, alumina reflector, core barrel, and B_4C neutron shield



Reactor vessel – release pathways

Release from fuel to reactor vessel

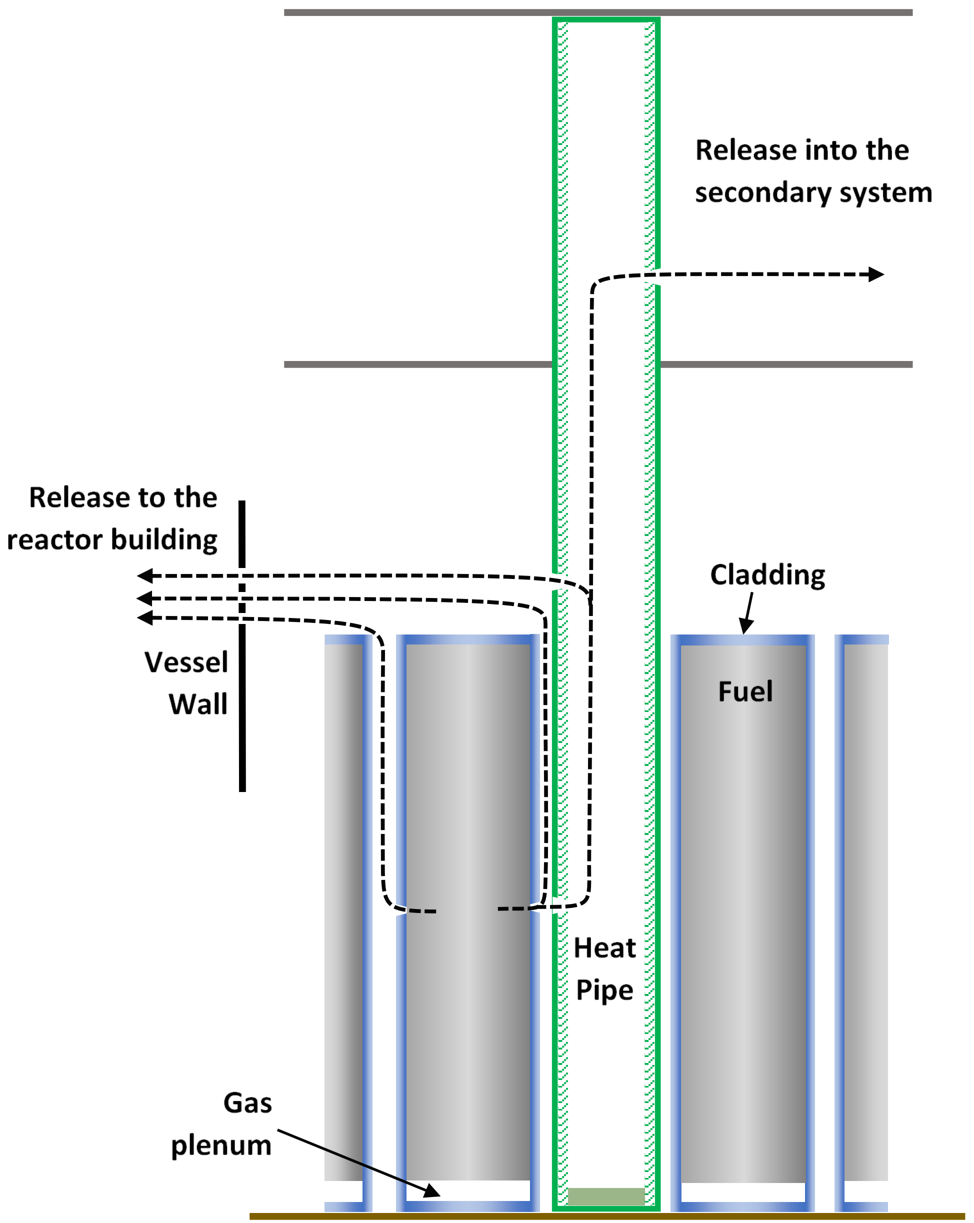
- Stainless-steel cladding failure at 1650 K

Release from reactor vessel to reactor building

- Assumed reactor vessel leakage

Heat-pipe release path

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



Enclosure building nodalization

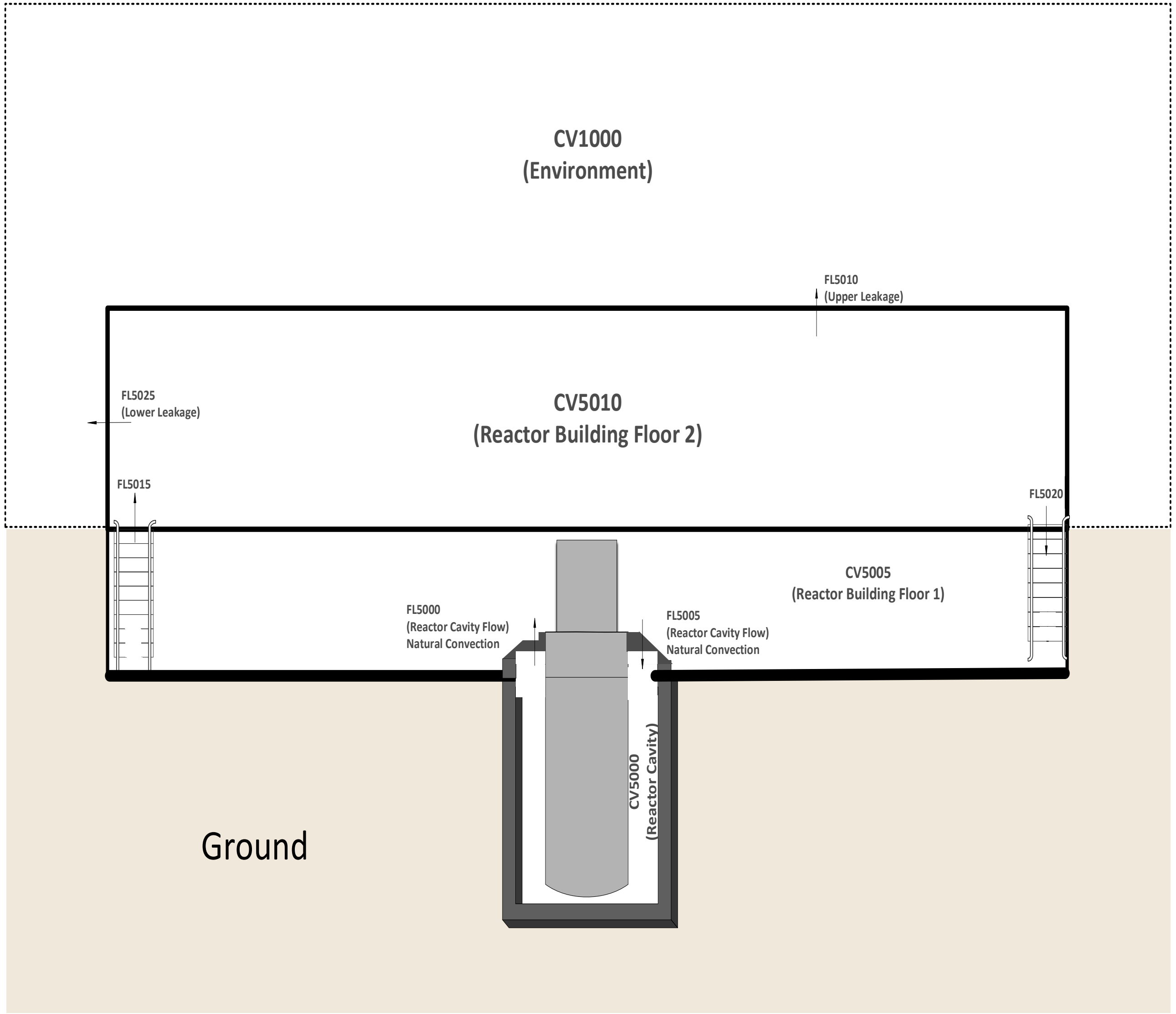
LANL and INL HPR descriptions did not address the enclosure building

Modeling includes internal building circulation flow paths

- Natural circulation into and out of the reactor cavity
- Natural circulation within the building

Building leakage addressed parametrically

- Base leakage similar to the reactor building surrounding the BWR Mark I containment



MELCOR input model attributes

Radionuclide inventory and decay heat at start of accident predicted with SCALE

- End-of-cycle inventory at 5-yr

Point kinetics model for transient power calculation

Heat transfer between adjacent fuel elements modeled using radiative exchange

- Heat transfer efficiency is parametrically varied

Potential for heat pipe creep rupture monitored in the evaporator and the condenser regions

Heat pipe limits estimated using LANL HTPIPE code

- MELCOR accepts sonic, capillary, entrainment and boiling limit curves
- Potassium and sodium limit curves were developed
- MELCOR can also accept proprietary performance curves when available

Description of the TOP scenario

Transient Overpower (TOP) scenario selected for demonstration calculations

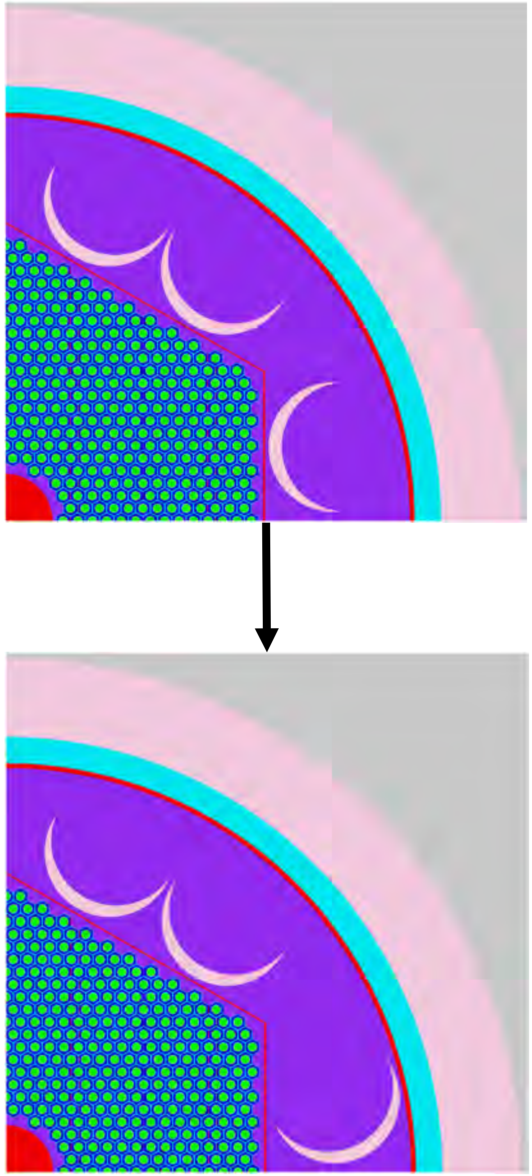
- Control drums malfunction and spuriously rotate “outward”

Modeled as linear reactivity insertion rate in $/second

- Safety control rods assumed to insert when peak fuel temperature exceeds 2200 K
- Strong feedback coefficient creates linear power increase

Performed sensitivity analysis to show how MELCOR could be used to gain insight into key source term drivers

- Sensitivities focused on source term and HPR parameters
- Previous LWR parameters do not necessarily translate to HPR uncertainties



Transient Overpower (TOP) scenario timeline

t = -5000 s

Steady-State

- Initialization
- Fuel temperature stabilizes

t = 0 s

Reactivity Insertion

- Power increase
- Temperature rise
- Heat pipe failure
- Core damage
- Fission product release

T_{max} = 2200 K

Post-SCRAM

- Radial cooling by natural processes
- Fission product release and transport

t = 24 h

Transient Overpower (TOP) base scenario (1/7)

The control drums start rotating at t=0 sec, which leads to an increase in the core power over 0.9 hr

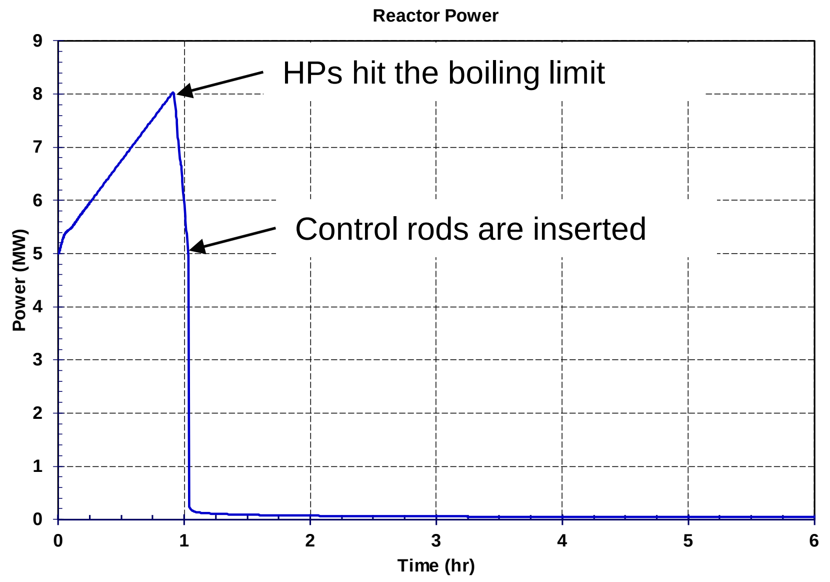
- Negative fuel temperature reactivity feedback limits the rate of power increase

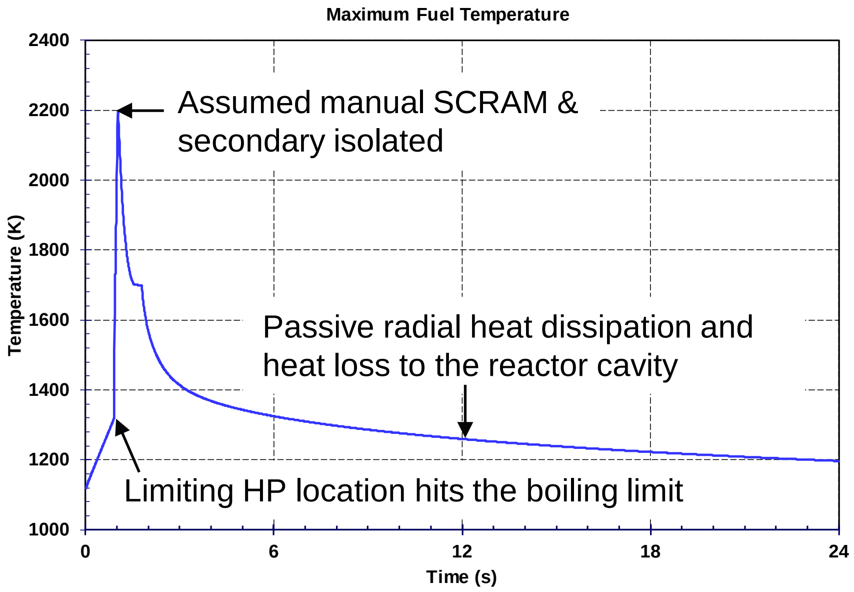
The core steadily heats until the maximum heat flux location reaches the boiling limit

- The heat transfer rate is limited above the boiling limit, which leads to a rapid heatup rate
- The SS cladding is assumed to fail at 1650 K (just below its melting point), which starts the fission product releases into the reactor
- The reactor is assumed to trip at 2200 K

Radial heat dissipation and heat loss to the reactor cavity passively cools the core

- No active heat removal (secondary system trips and isolates)



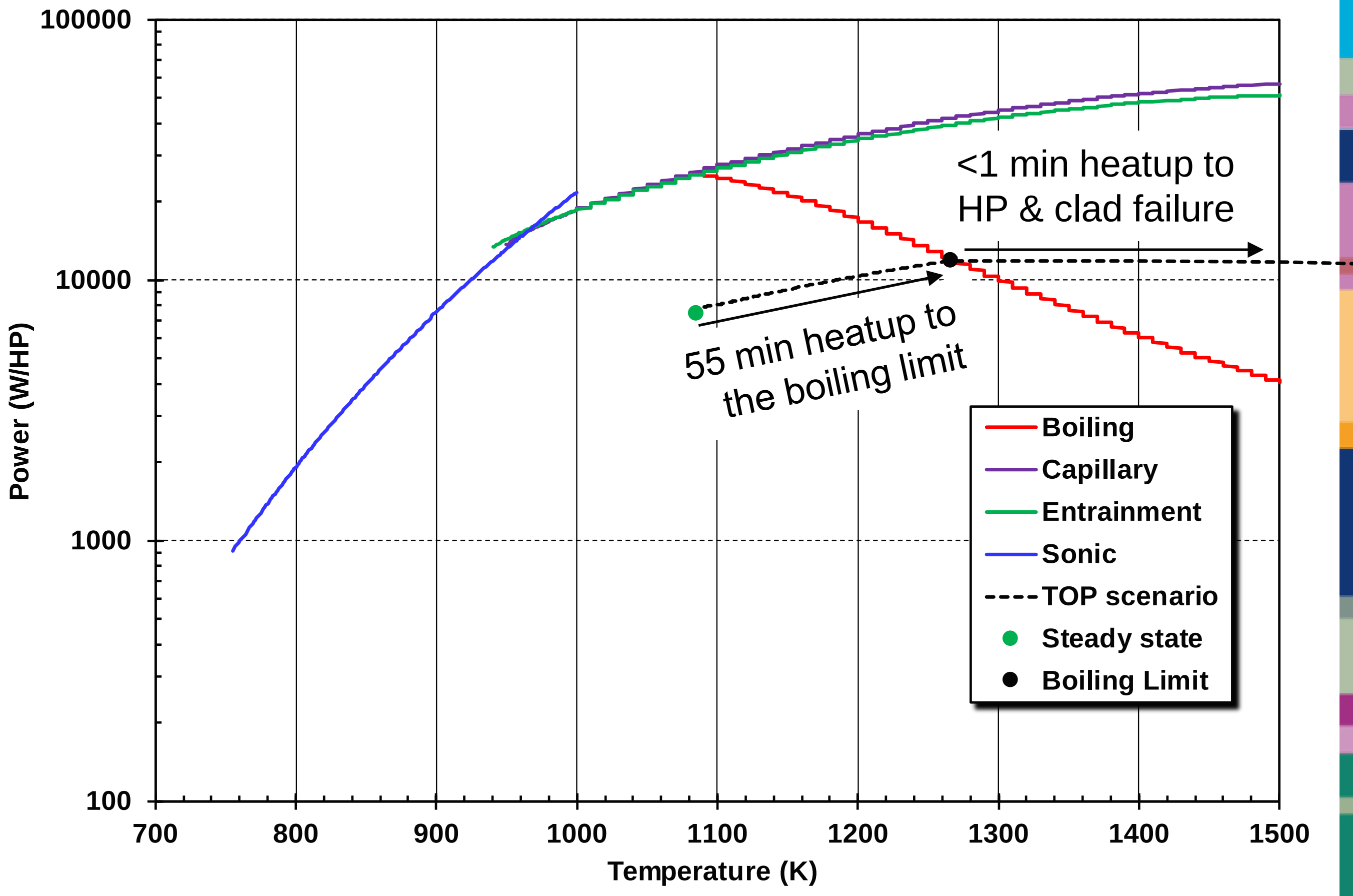


Transient Overpower (TOP) base scenario (2/7)

The HP performance limits at the highest heat flux location show a steady heatup to the boiling limit

- Once the boiling limit is reached, there is a rapid heatup over the next minute
  - The fuel rapidly heats to melting conditions
  - SS cladding fails at 1650 K
  - SS HP wall also fails at 1650 K
- The start of the fission product release occurs through the failed cladding locations

HP performance limit curves with the TOP response



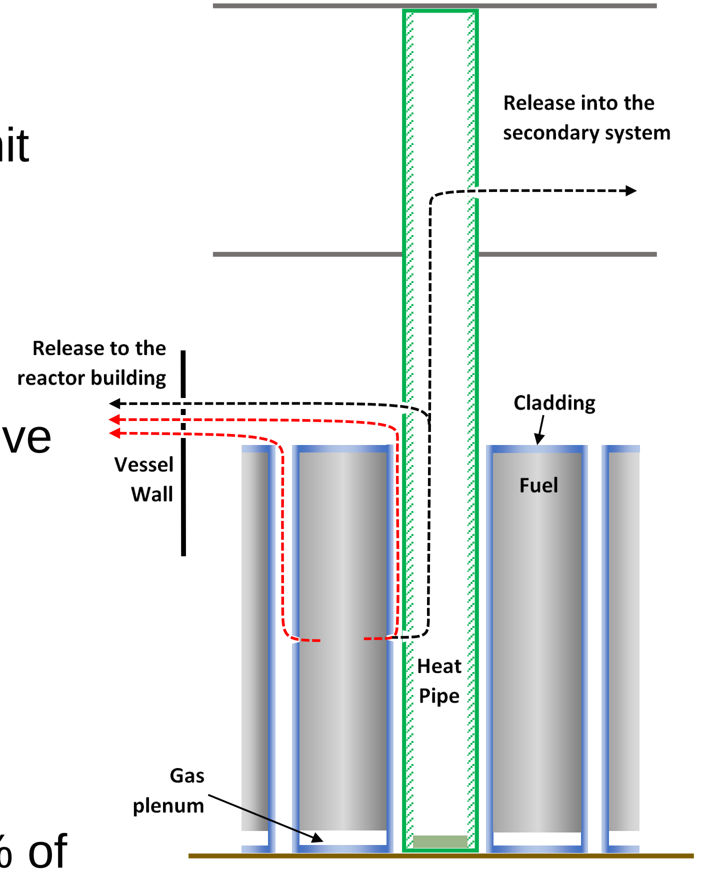
Transient Overpower (TOP) base scenario (3/7)

Cladding failure at 1650 K resulting in fission product release

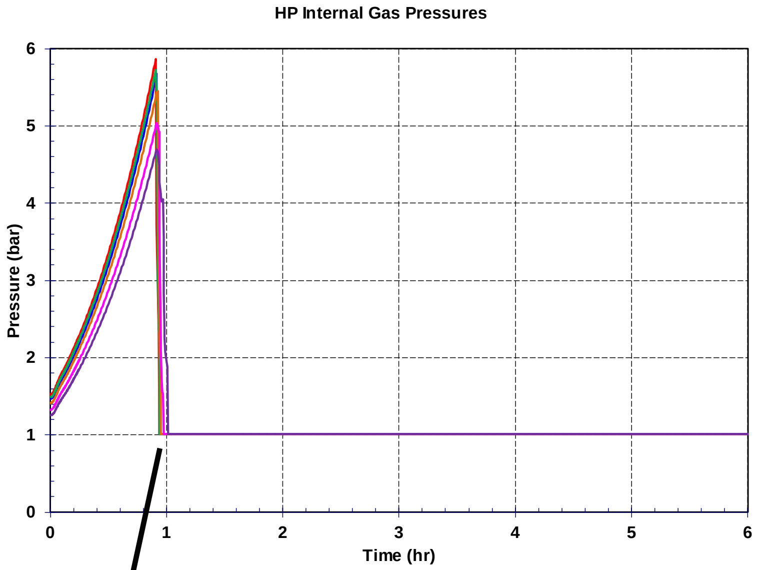
- HPs that exceeded the boiling limit rapidly heat to cladding failure (1650 K)
- ~20% of the 1134 HPs and fuel elements failed
- HP depressurization on failure drive release from the vessel

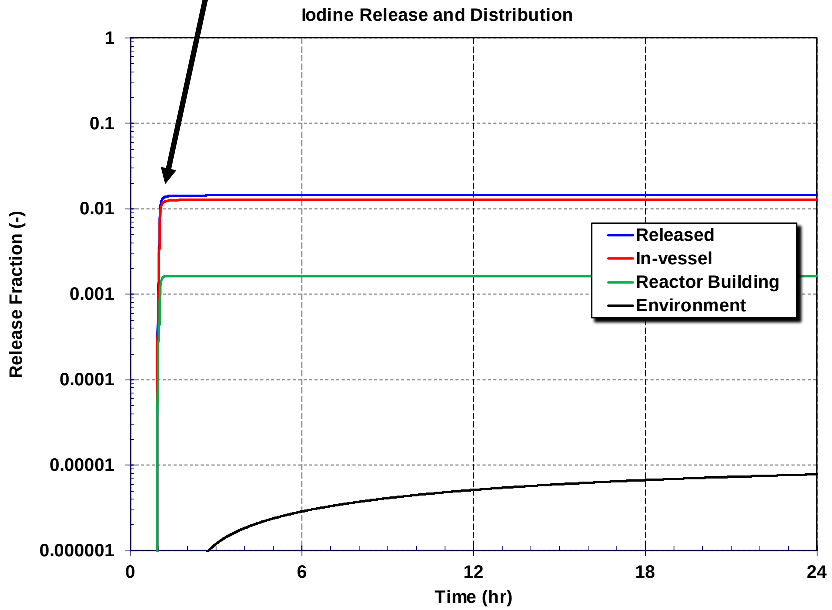
Iodine releases also depend on time at temperature

- Fuel release – 1.4% of core inventory
- Environmental release – 0.0008% of core inventory



- Vessel leakage is 1.6 in^2
- Building leakage is 1.8 in^2





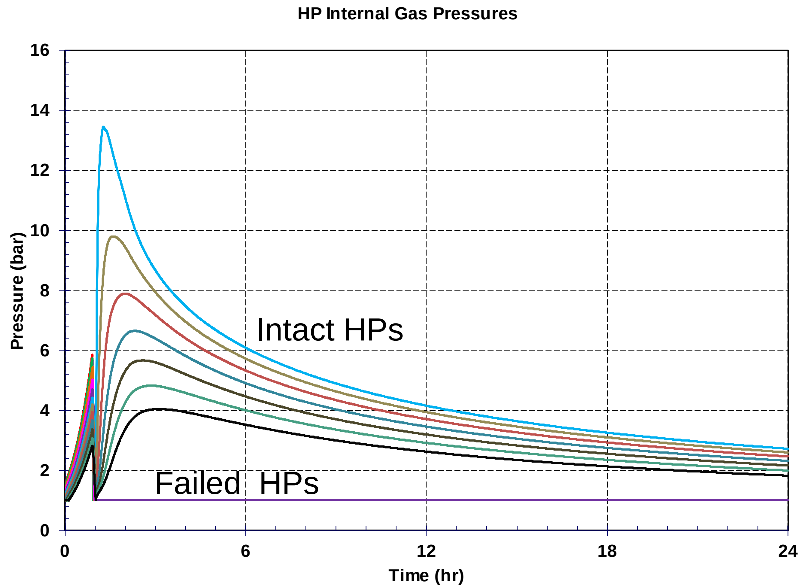
Transient Overpower (TOP) base scenario (4/7)

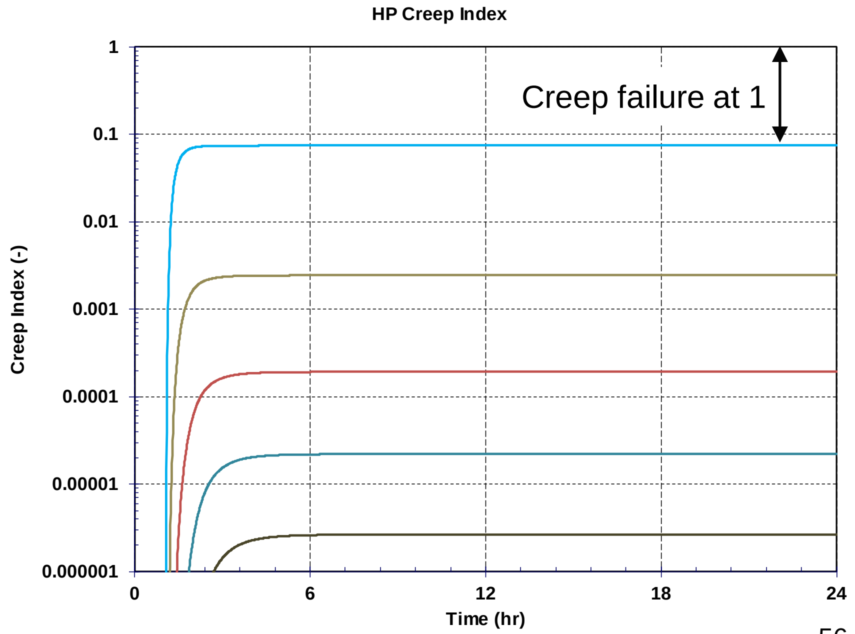
The HPs could be challenged by creep failure at high temperature and pressure

- The HP gas heats and pressurizes during the TOP scenario
- The HP depressurizes after the wall fails shortly after reaching the boiling limit
  - Creep accumulation effectively stops upon HP wall failure without ΔP stress
- For HPs that do not reach the boiling limit, the HP pressure initially drops due to secondary system removing heat

HP creep failure is monitored using Larson-Miller correlations

- TOP base scenario shows maximum creep is ~0.07 (failure = 1)
- Creep failures in the condenser can create a bypass leak path to the environment

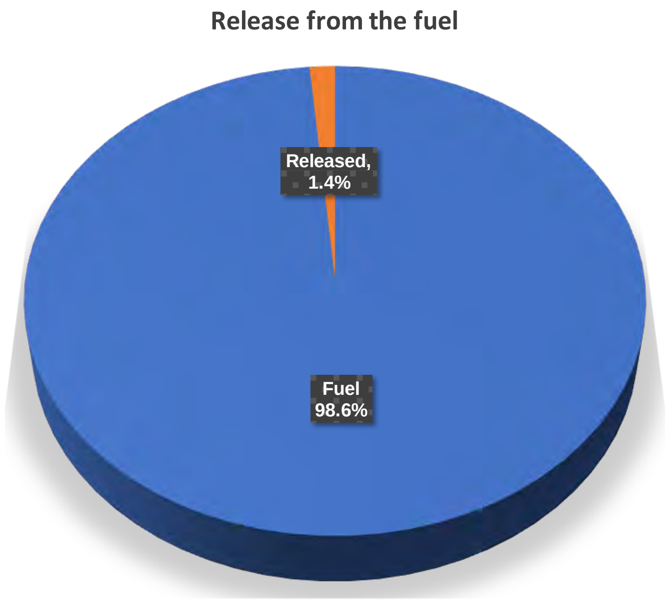




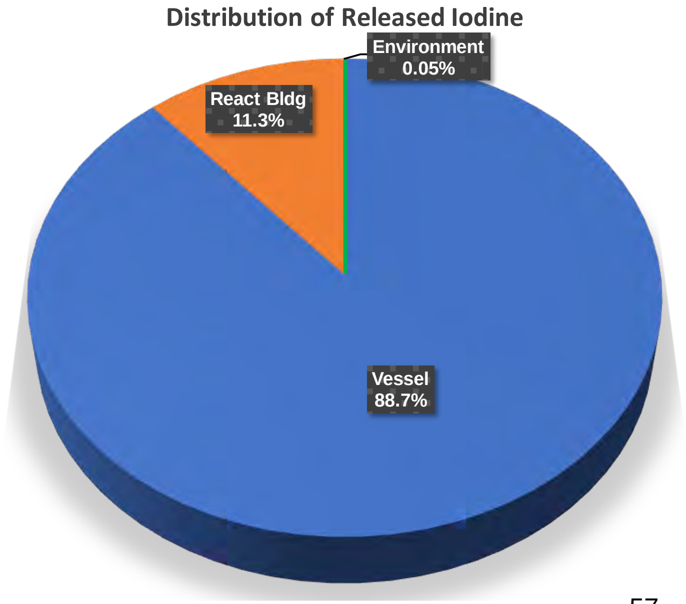
Transient Overpower (TOP) base scenario (5/7)

Fission products are retained in the fuel or deposit on their way to the environment

- The cladding remained intact for ~80% of the fuel elements
- 98.4% of the iodine fission product inventory is retained in the fuel due to limited time at high temperature



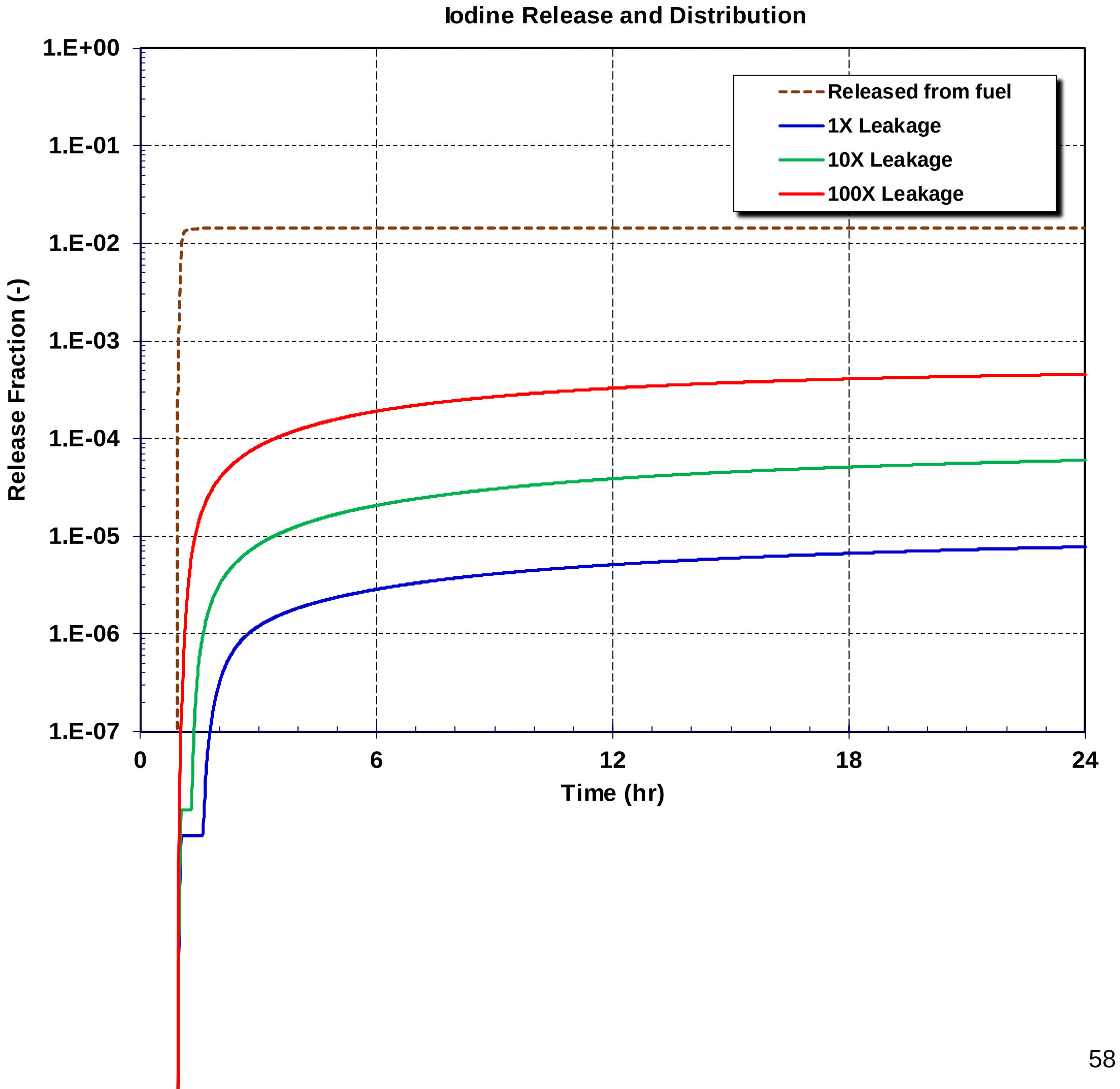
- The vessel retains 89% of the released iodine radionuclides
  - HP depressurization after failure is primary release mechanism
- The reactor building retains 11% of the radionuclides in the base case
  - BWR reactor building leak tightness used for the base case
  - No strong driving pressure to cause leakage



Transient Overpower (TOP) base scenario (6/7)

A series of calculations were performed to investigate the sensitivity of the source term magnitude to reactor building leakage effects

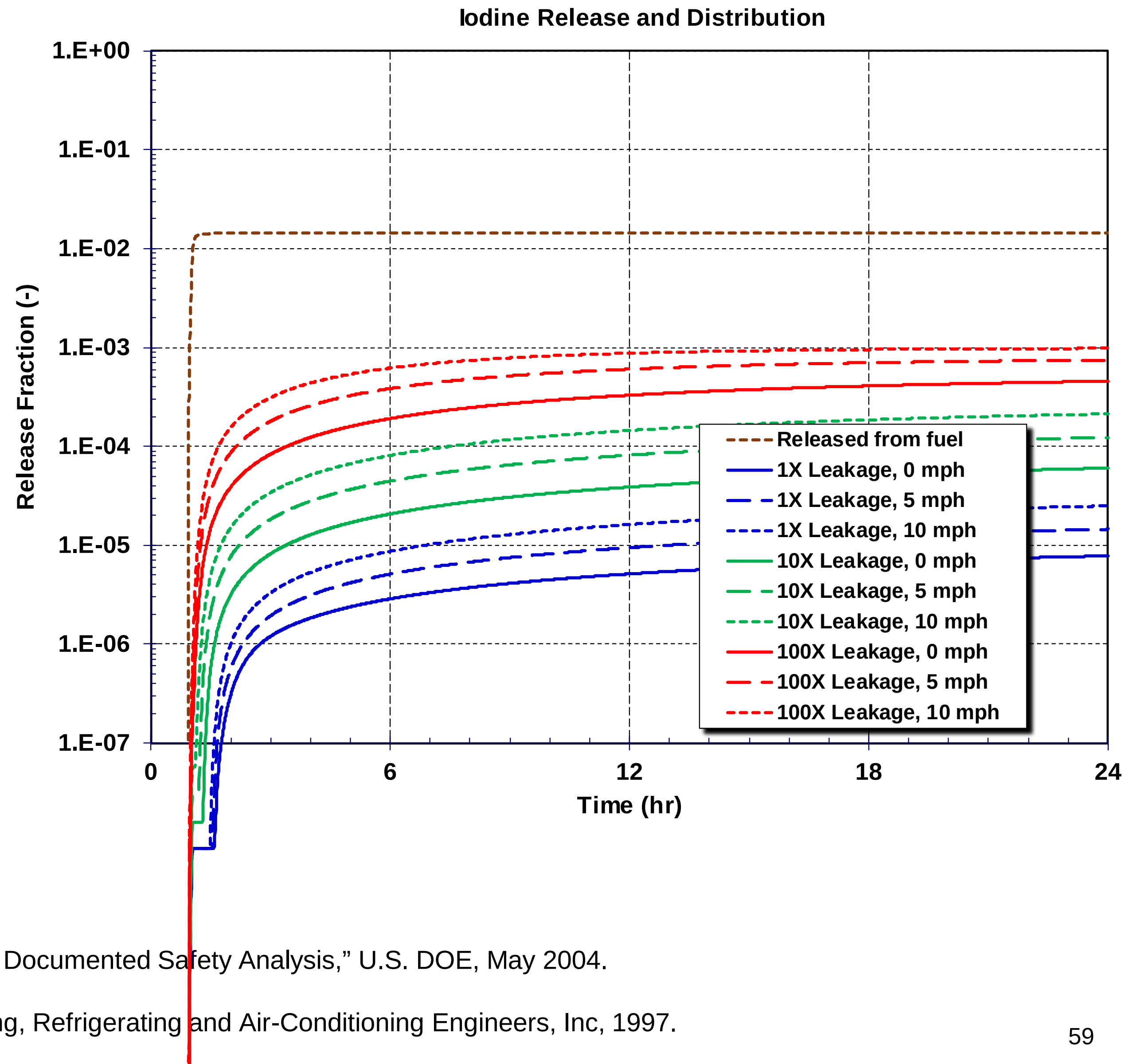
- The design specifications of the reactor building were assumed
  - The base result (1X) assumed a BWR reactor building value
  - 10X and 100X reflects higher design leakage and/or building damage
- Building leakage is driven by a very small temperature gradient to the environment (~5-7 °C)
  - Leakage is approximately linear with leakage area (1X is ~1.8 in^2)



Transient Overpower (TOP) base scenario (7/7)

A series of calculations were performed to investigate the impact of an external wind

- External wind effects are included in DOE facility safety analysis where there also are not strong driving forces
  - Wind increases building infiltration and exfiltration
  - Upwind and downwind leakage pathways
- Wind effects are modeled as a Bernoulli term
  - $dP = \frac{1}{2}\rho C_p v^2$
  - ASHRAE building wind-pressure coefficients



External wind modeling ref:

“MELCOR Computer Code Application Guidance for Leak Path Factor in Documented Safety Analysis,” U.S. DOE, May 2004.

Building wind pressure coefficients.

ASHRAE, 1977, Handbook of Fundamentals, American Society of Heating, Refrigerating and Air-Conditioning Engineers, Inc, 1997.

Heat Pipe Reactor MELCOR Uncertainty Analysis

Role of MELCOR in Resolving Uncertainty

Uncertainty

Event Scenario Uncertainty

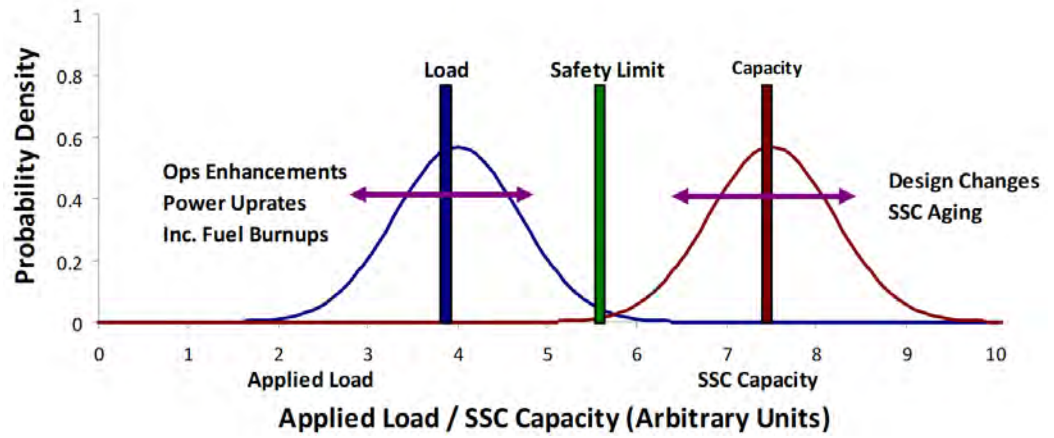
Phenomenological Model Uncertainty

Plant Initial/Boundary Condition Uncertainty

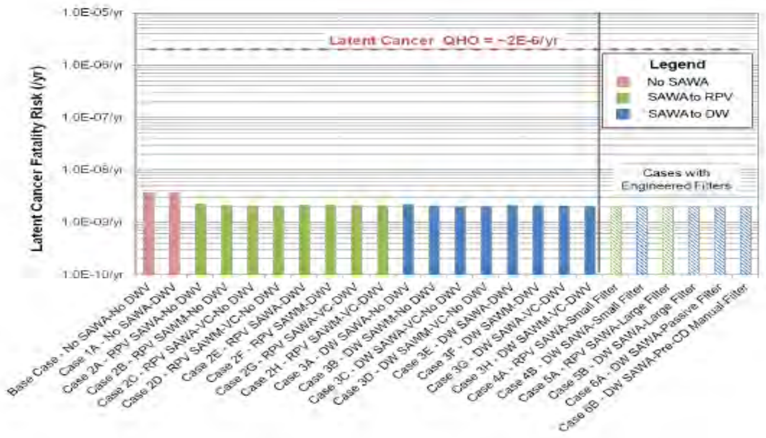
SSC Failure Modes

Simulation Uncertainty

Engineering Performance



Risk-Informed Assessment



Evolution from MELCOR LWR Uncertainty Analysis

MELCOR application to LWR severe accident uncertainties

- Range of uncertainty studies under SOARCA
- PWR and BWR plant uncertainty studies
- Resolved role of uncertainty in a number of critical severe accident issues of high impact

General commonalities between LWR and HPR accident uncertainties

- Chemical form of key elements
- Aerosol physics parameters (e.g., shape factor)
- Operating time before accident happens
- Containment leakage hole size

Parameter selection emphasized potential HPR-specific uncertainties

- Ran samples of uncertainty calculations to explore role of uncertainty in evolution of HPR accident scenario class

Parametric Uncertainties – Capability Demonstration

Component	Parameter	Ranges
Heat Pipes	Heat Pipe Failure Location	Condenser (50%) / Evaporator (50%)
	Initial non-functional HPs	0% - 5%
Core	Gaseous Iodine Fraction (-)	0.0 - 0.05
	Reactivity Insertion Rate ($/s)	0.5×10^{-4} - 1.0×10^{-3}
	Total reactivity feedback	-0.0015 to -0.0025
Vessel	Fuel Element Radial View Factor Multiplier (-)	0.5 - 2.0
	Vessel Emissivity (-)	0.125 - 0.375
	Total Leak Area (m^2)	2×10^{-5} - 2×10^{-3}
	Vessel and Vessel Upper Head HTC (W/m-K)	1 – 10
Confinement	Cavity entrance open fraction	100% (90%) - 1% (10%)
	Cavity Emissivity (-)	0.125 – 0.375
	Wind Loading (m/s)	0 – 10
	Total Leak Area Multiplier (-)	1 - 100
Scenario	Peak fuel temperature for safety rod insertion (K)	1300 – 2200

Characterization of Uncertainty in Event Evolution

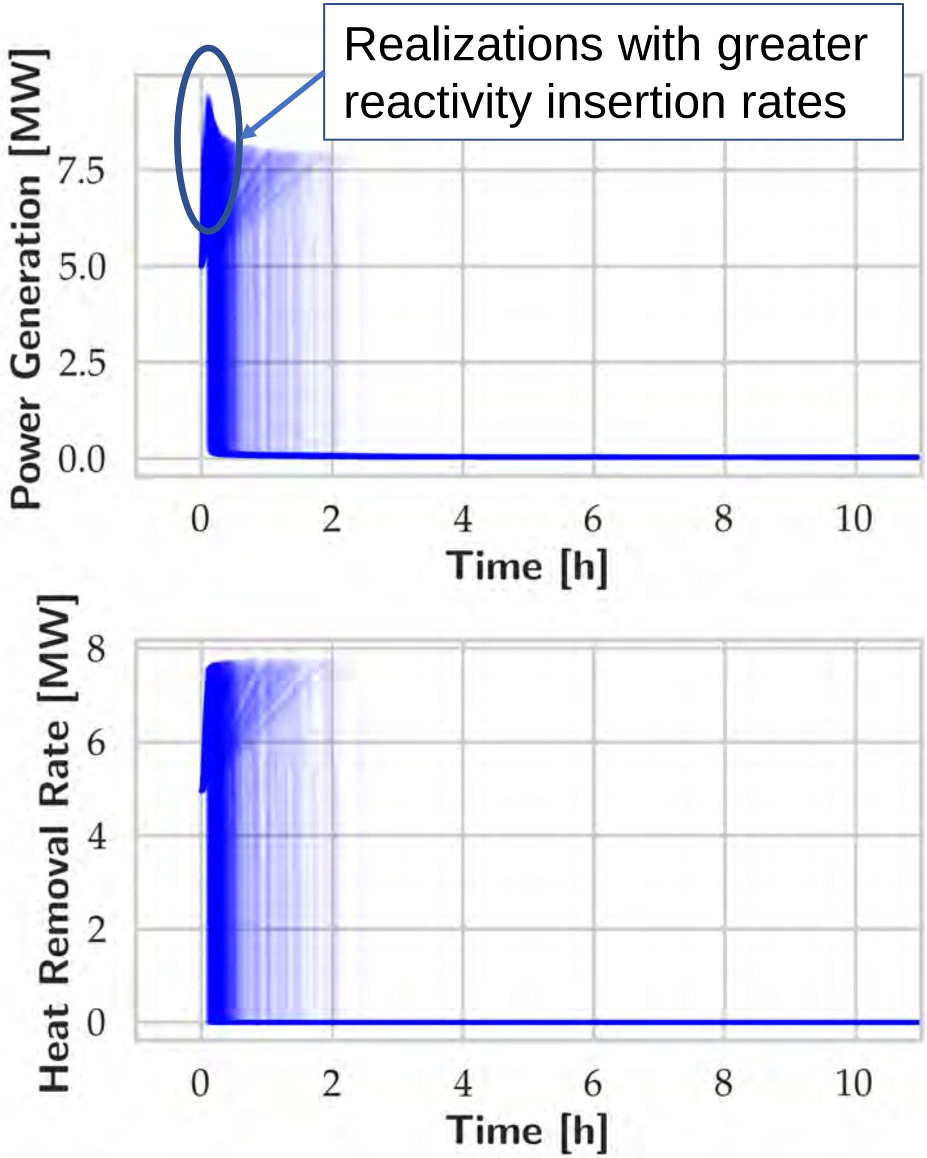
Traditional event scenario evolution for LWRs dominated by active system performance

Event scenarios evolved based often on binary decisions

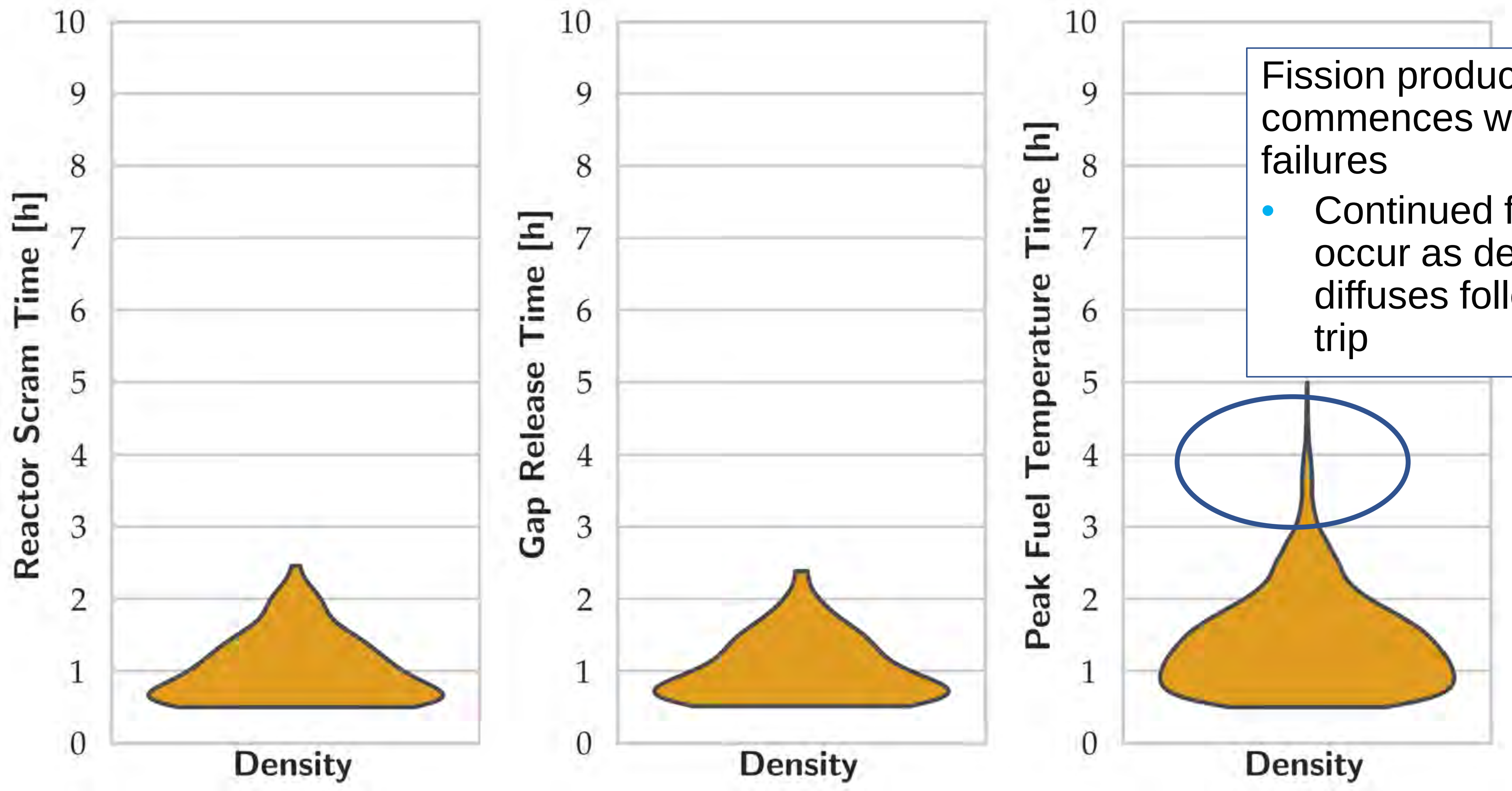
- SSC performance often characterized as success or failure
- Risk profile could be adequately characterized or bounded by success or failure of SSCs

HPR accident scenario evolution will be unique, like other advanced non-LWRs

- Limited operational experience
- Broader range of operation for passive systems
- Consideration of degraded modes of operation
- What is the true margin to failure under accident conditions?



Overall Timing of Event Evolution



Fission product release commences with cladding failures

- Continued fuel heatup can occur as deposited energy diffuses following reactor trip

Evaluating Heat Pipe Response

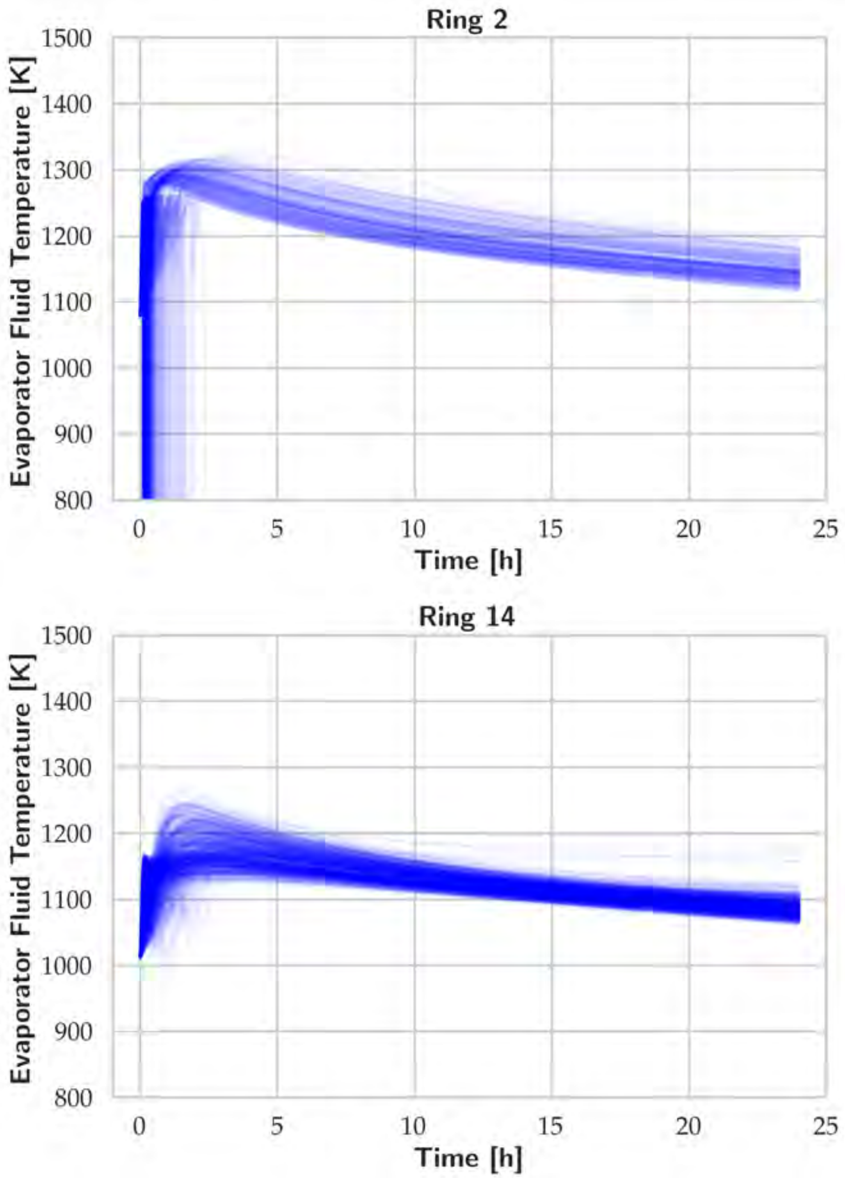
Spectrum of accident scenarios give rise to range of plant conditions

- Relevant to assessing potential and magnitude of consequences

Evaluation of SSC performance and margin in performance under accident conditions

HPRs rely on passive heat removal through capillary flows in heat pipes

- Sensitive to operating range of heat pipes
- Operating limits could for example be challenged under overpower conditions



Fuel Response by Ring

Highest powered rings off-center

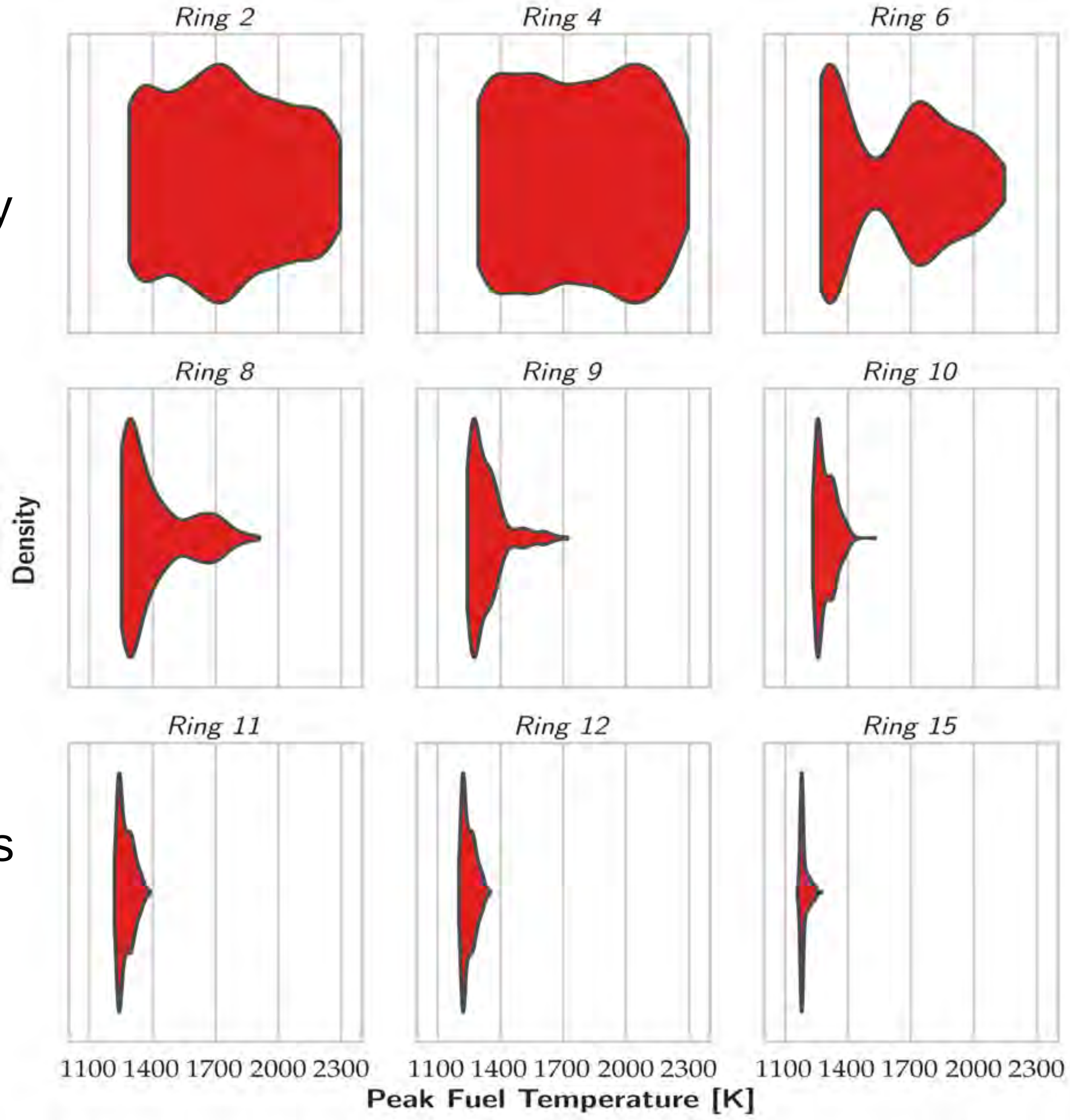
Energy deposited in reactor during reactivity transient diffuses to lower power rings after reactor trip

Heatup of fuel in peripheral rings influenced by

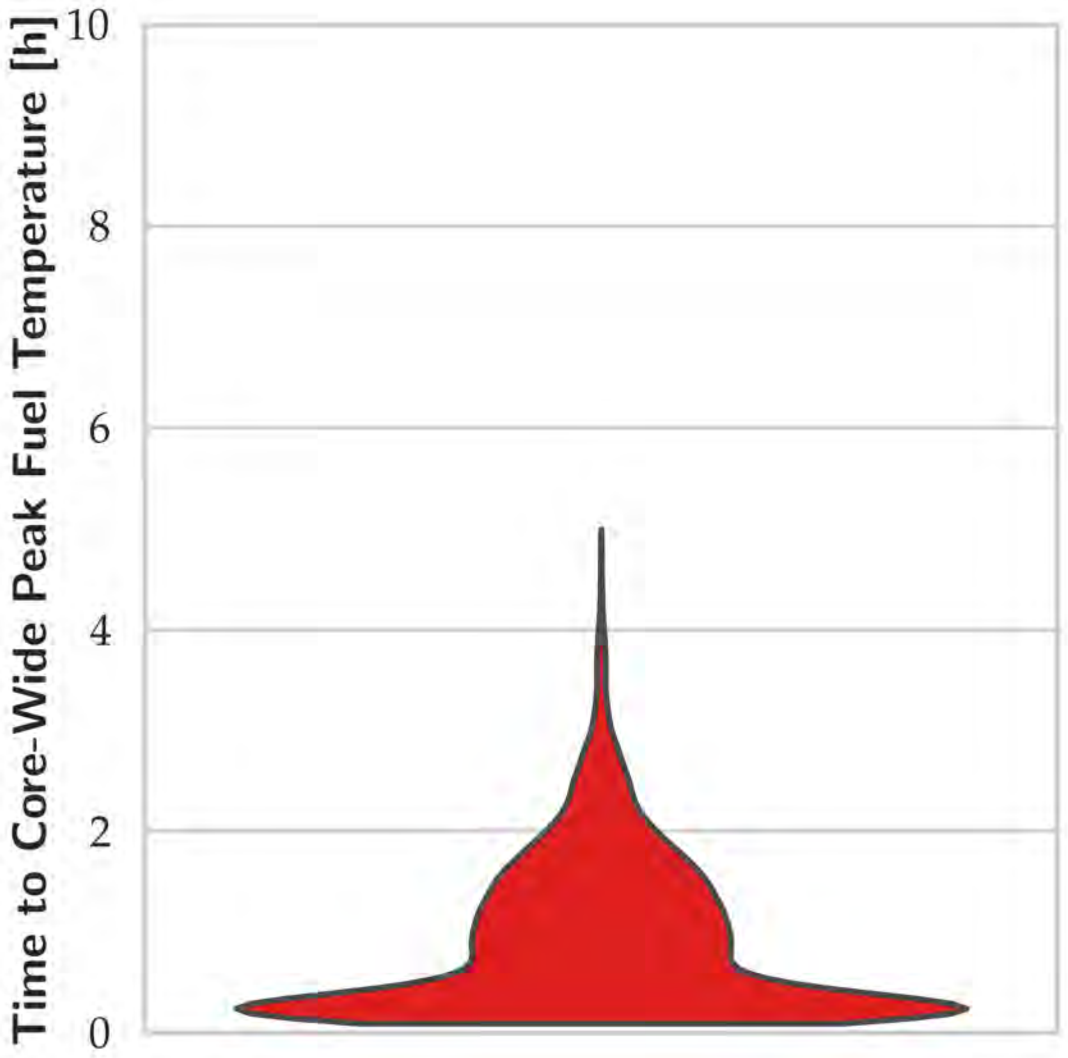
- Lower decay heat levels
- Energy loss to confinement through vessel wall

Heatup of fuel in central rings influenced by

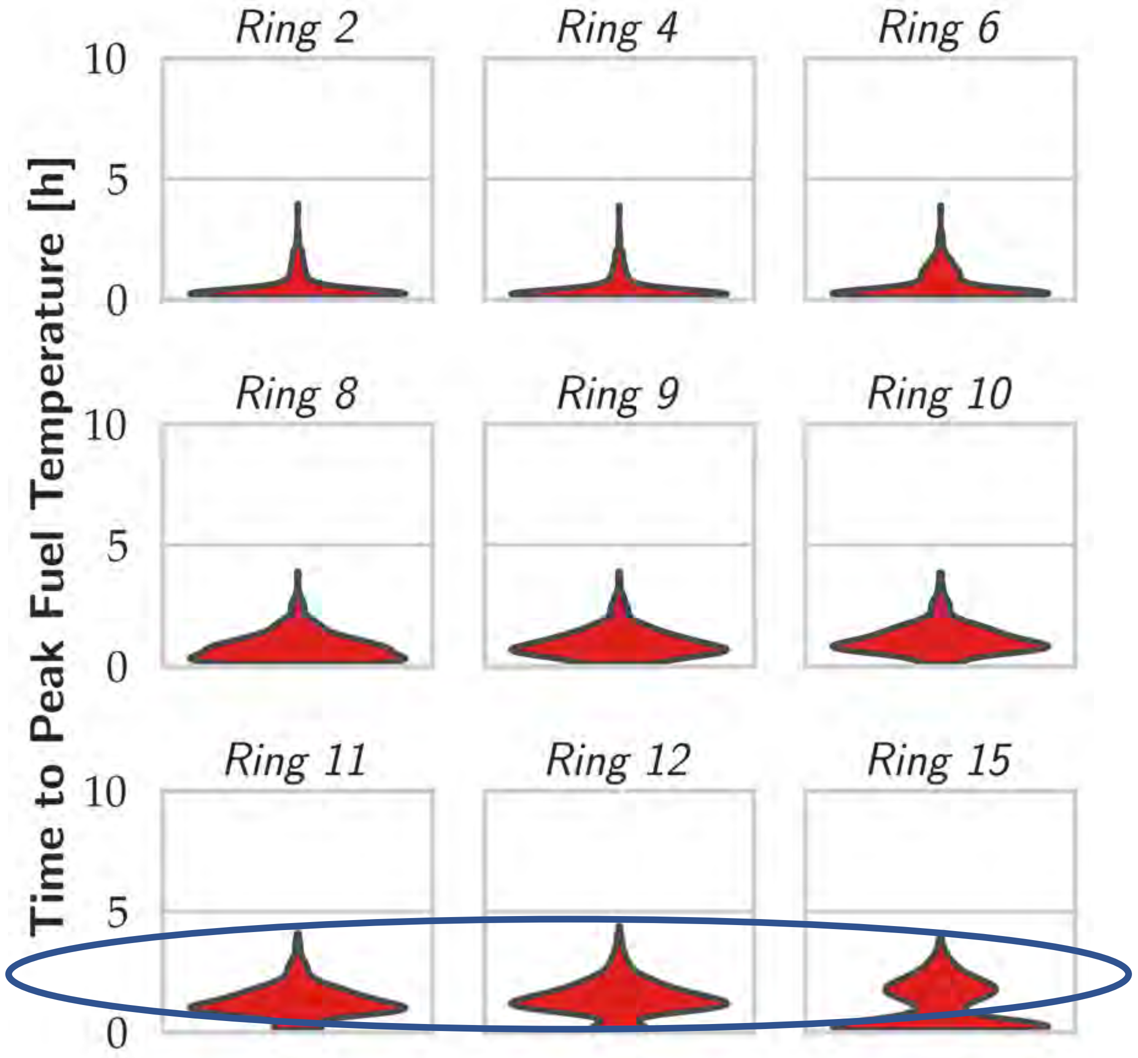
- Diffusion of energy from hottest fuel rings
- Limited heat sinks to which to dissipate energy



Thermal Inertia in Fuel Response



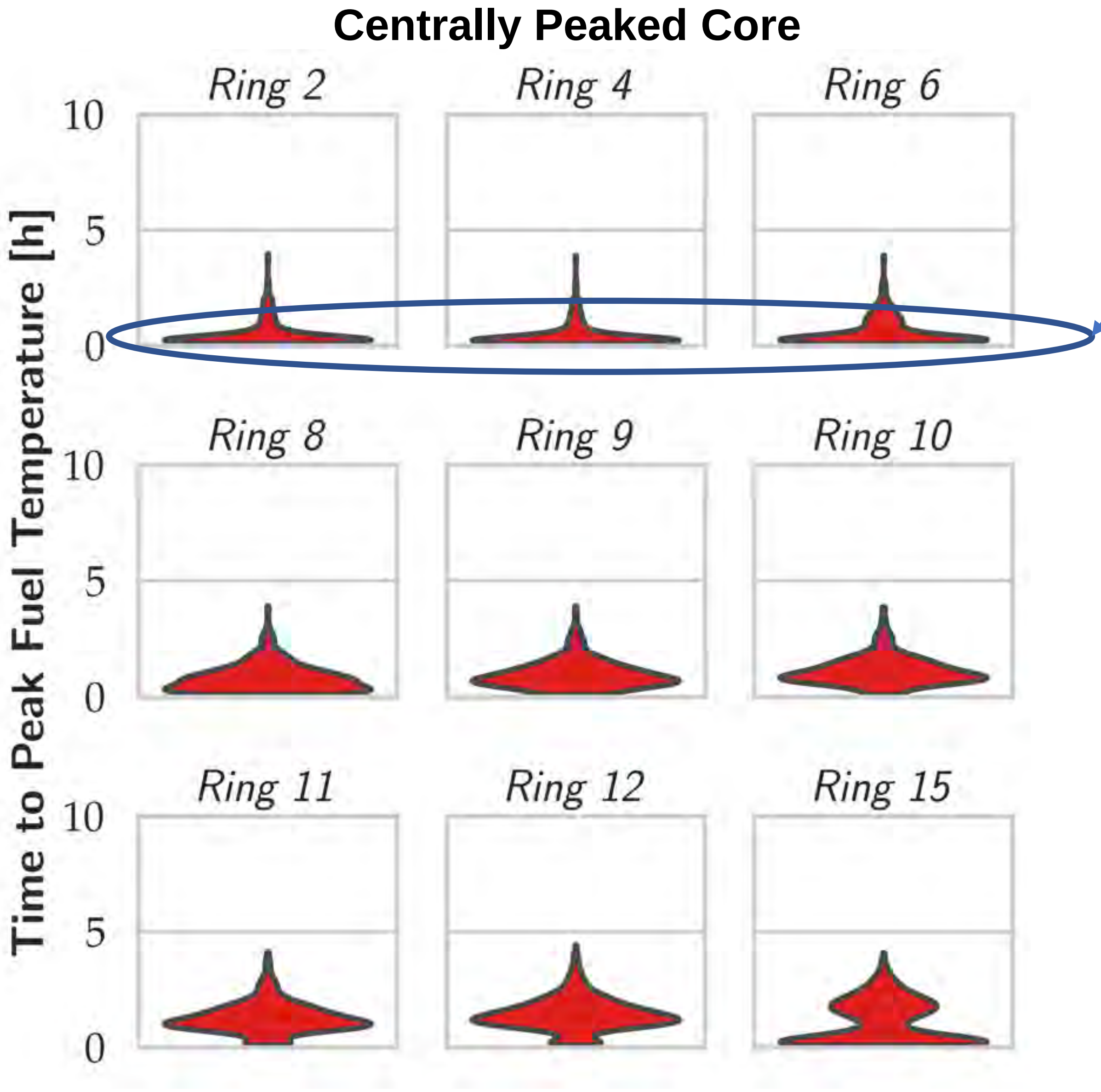
Most realizations dominated by early energy deposition into fuel prior to reactor trip

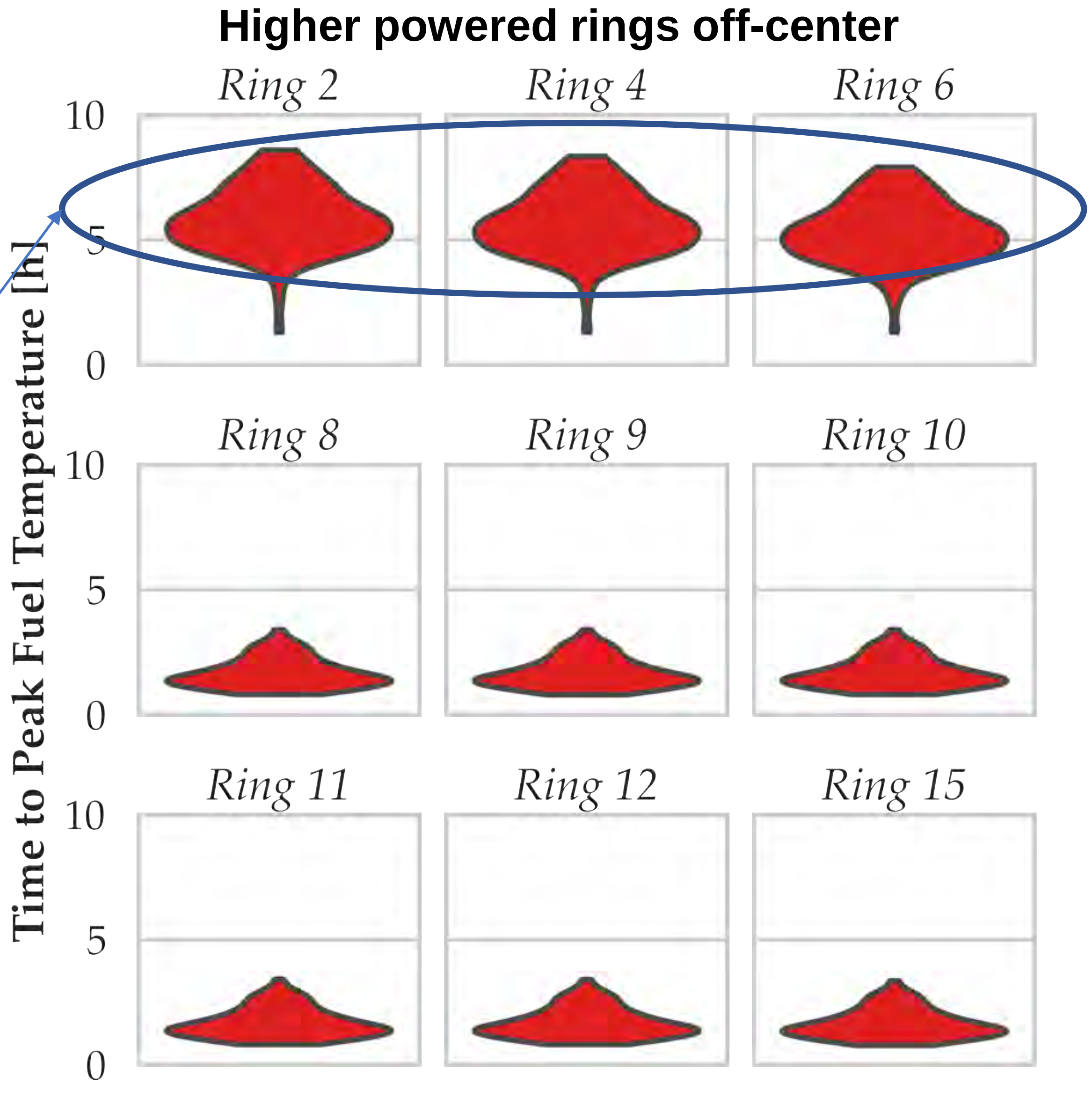


Diffusive heat flux from hottest rings to periphery

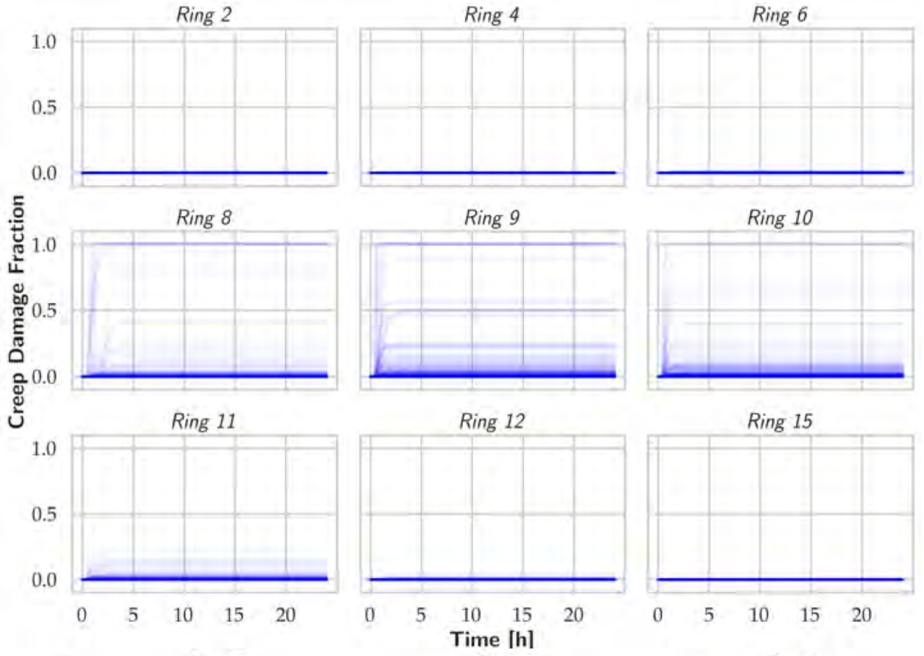
- Dominates heatup of fuel in peripheral rings

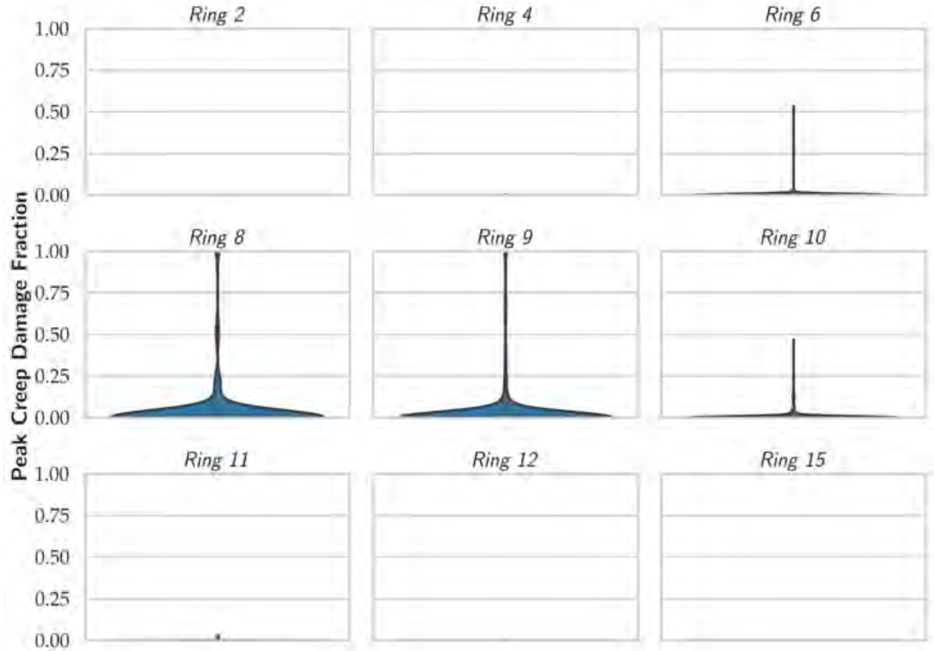
Thermal Inertia in Fuel Response

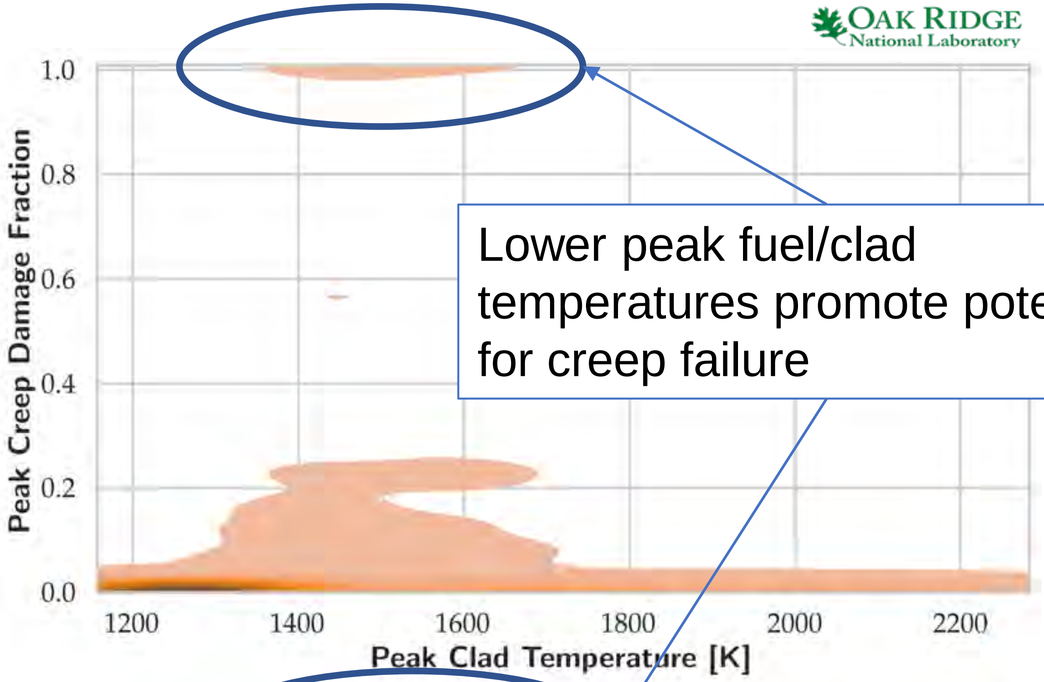




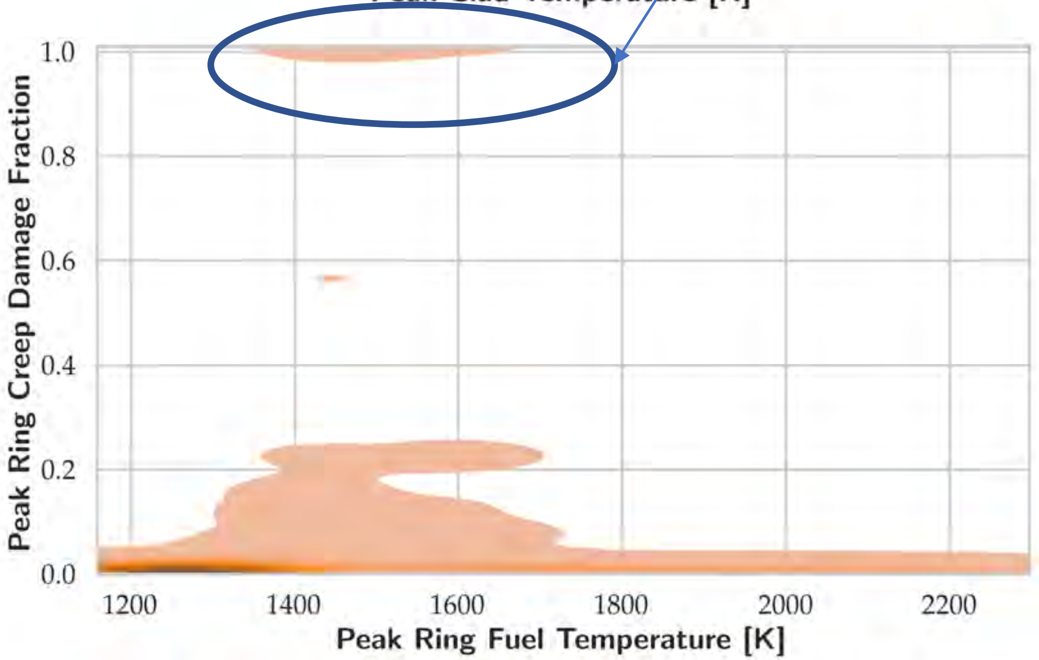
Heat Pipe Response



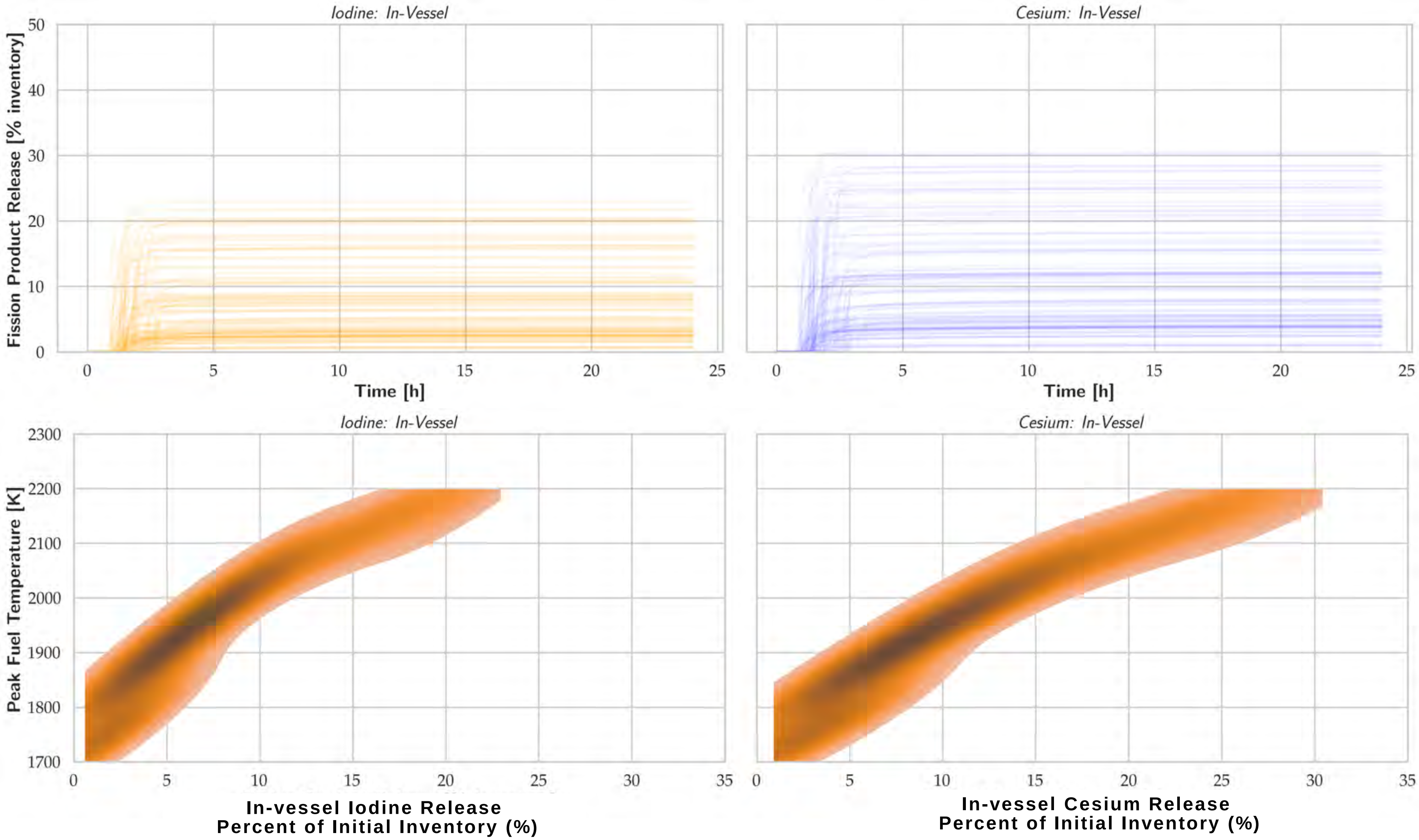




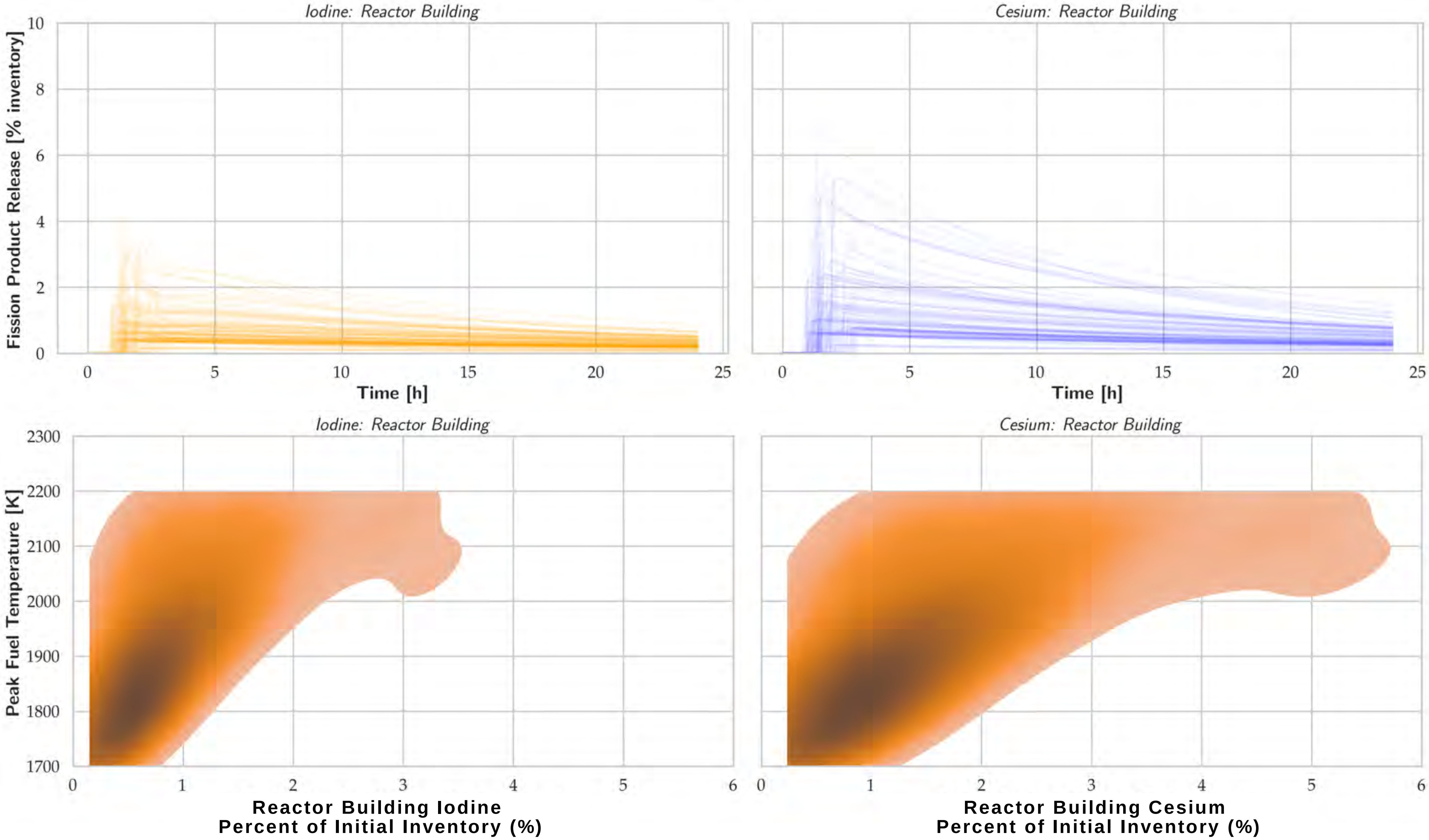
Lower peak fuel/clad temperatures promote potential for creep failure



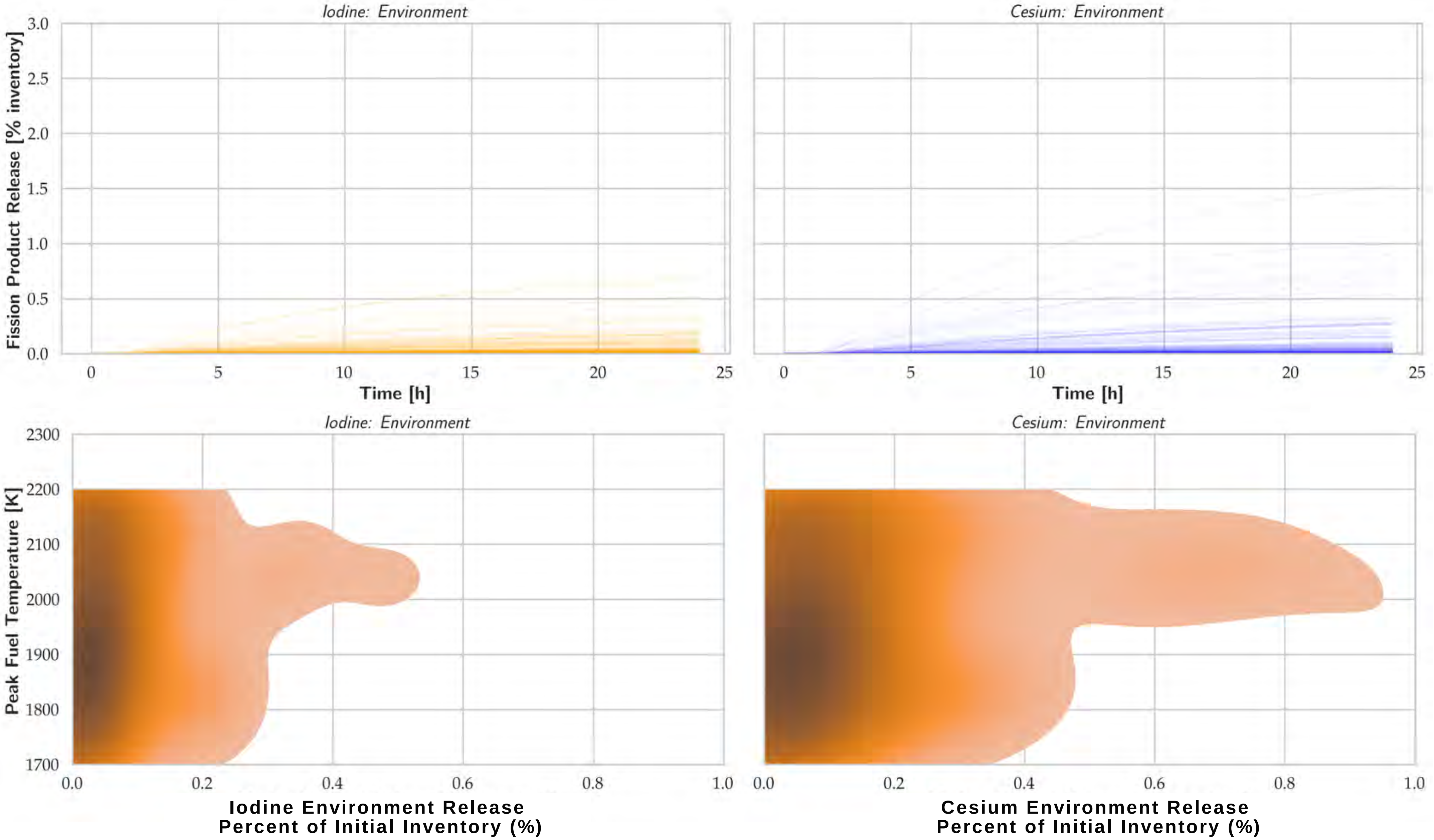
Fission Product Release from Fuel Characterization



Fission Product Transport Characterization



Fission Product Release to Environment



Summary

Conclusions

Added HPR modeling capabilities to SCALE & MELCOR for HPR source term analysis to show code readiness

Modeling demonstrated for a Transient Overpower Scenario with delayed scram

- Input of detailed ORIGEN radionuclide inventory data from ORNL
- Input radial and axial power distributions from ORNL neutronic analysis
- Develop MELCOR input model for exploratory analysis
- Fast-running calculations facilitate sensitivity evaluations (600 realizations included in the exploratory calculations)

Developed an understanding of non-LWR beyond-design-basis-accident behavior and overall plant response

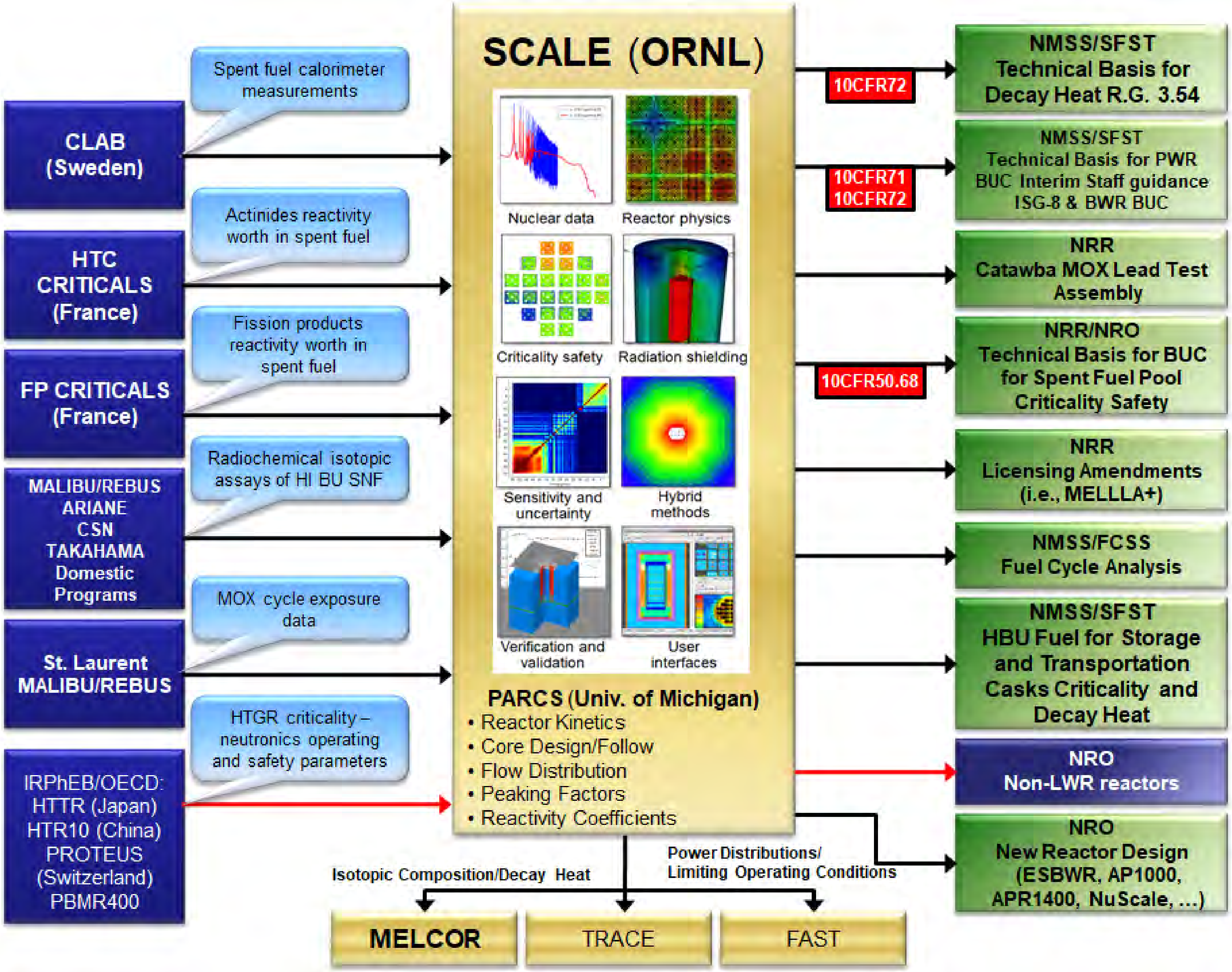
SCALE Overview

U.S.NRC

Oak Ridge National Laboratory

Sandia National Laboratories

SCALE Development for Regulatory Applications



What Is It?
The SCALE code system is a modeling and simulation suite for nuclear safety analysis and design. It is a modernized code with a long history of application in the regulatory process.

How Is It Used?
SCALE is used to support licensing activities in NRR (e.g., analysis of spent fuel pool criticality, generating nuclear physics and decay heat parameters for design basis accident analysis) and NMSS (e.g., review of consolidated interim storage facilities, burnup credit).

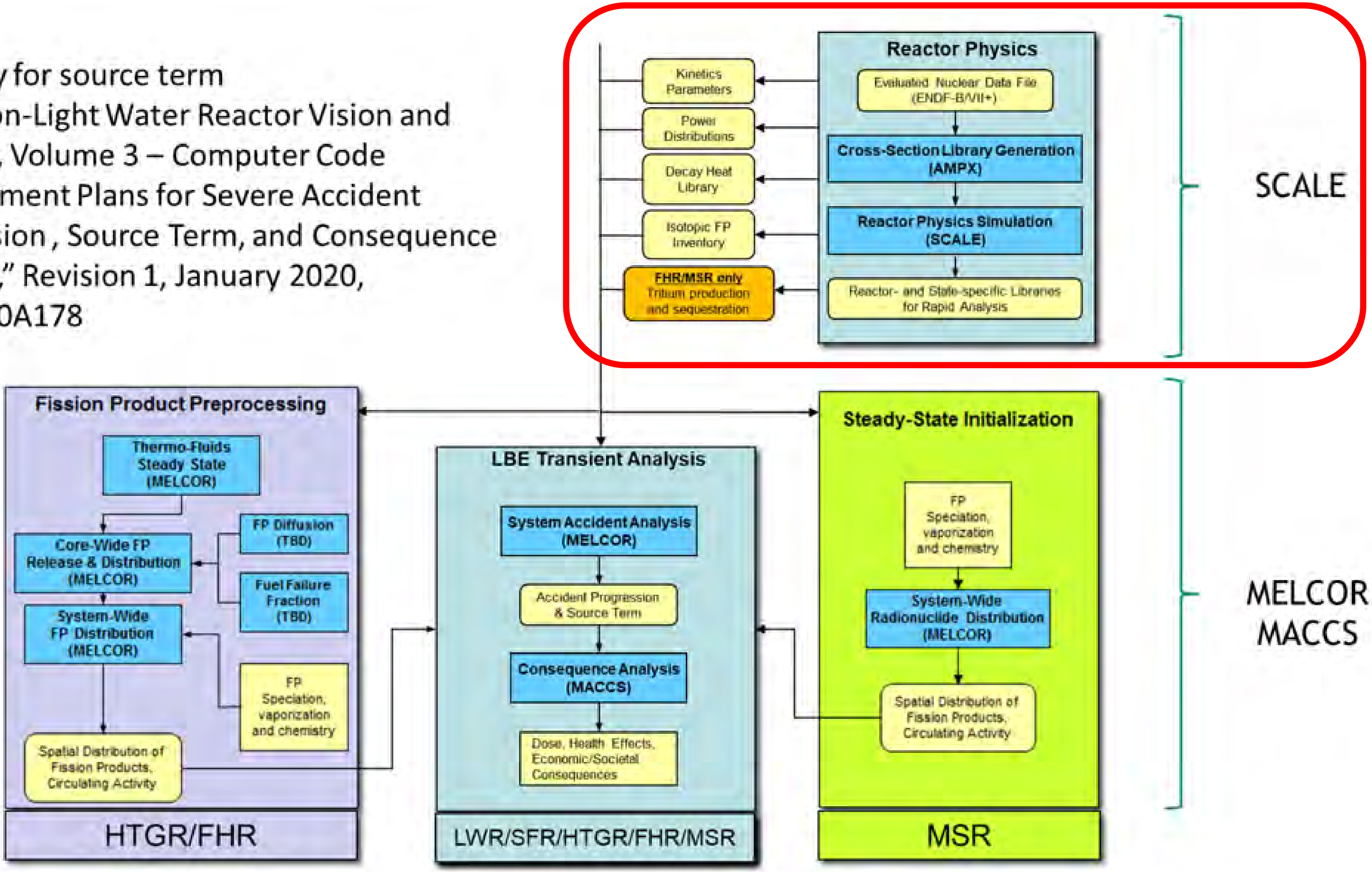
Who Uses It?
SCALE is used by the U.S. Nuclear Regulatory Commission (NRC) and in 61 countries (about 10,000 users and 33 regulatory bodies).

How Has It Been Assessed?
SCALE has been validated against criticality benchmarks (>1000), destructive assay of fuel and decay heat for PWRs and BWRs (>200)

Data to generate for MELCOR: QOIs

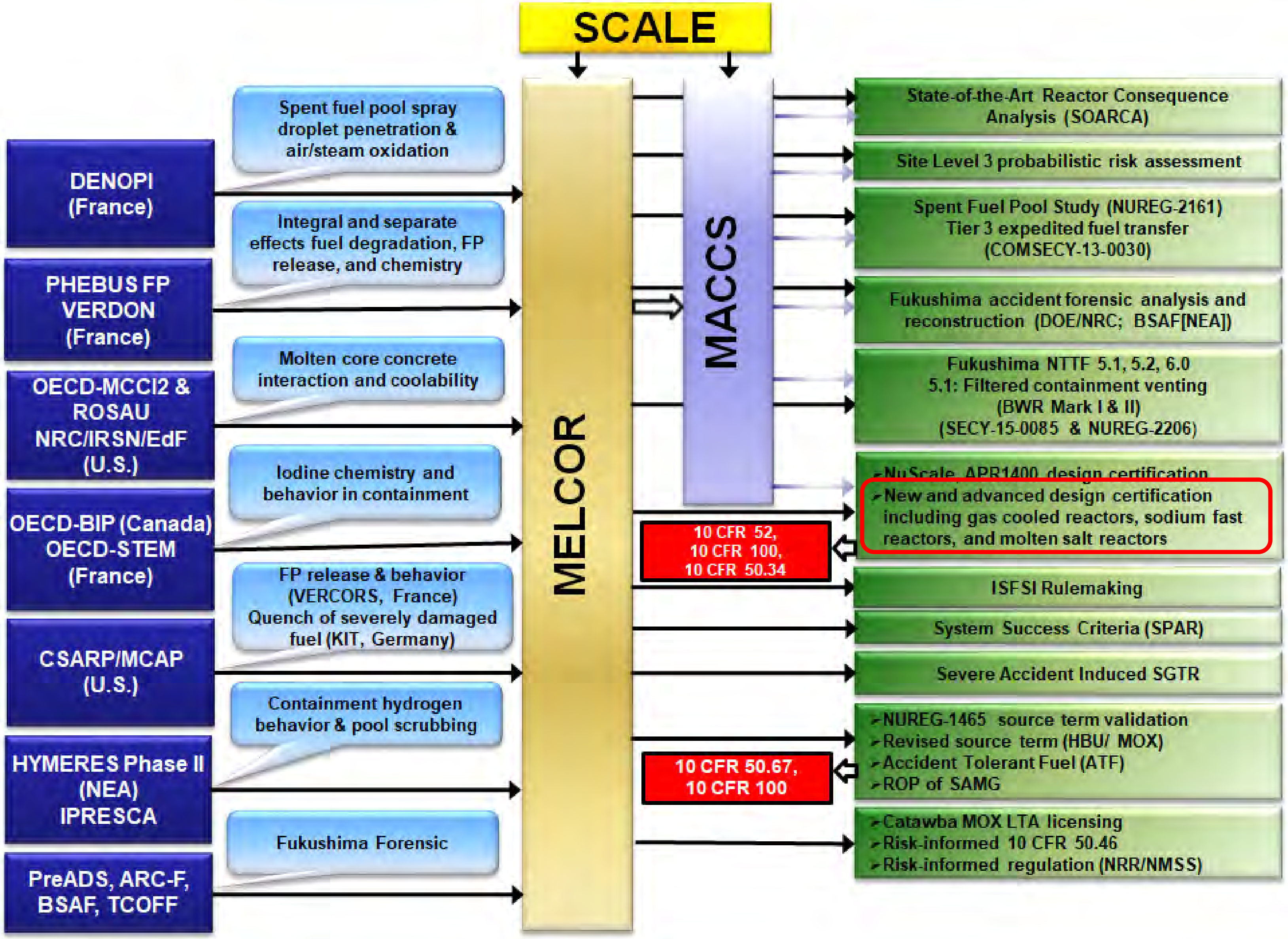
Code strategy for source term

"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 for Accident Progression and Source Term Analysis

MELCOR Development for Regulatory Applications



What Is It?
MELCOR is an engineering-level code that simulates the response of the reactor core, primary coolant system, containment, and surrounding buildings to a severe accident.

Who Uses It?
MELCOR is used by domestic universities and national laboratories, and international organizations in around 30 countries. It is distributed as part of NRC's Cooperative Severe Accident Research Program (CSARP).

How Is It Used?
MELCOR is used to support severe accident and source term activities at NRC, including the development of regulatory source terms for LWRs, analysis of success criteria for probabilistic risk assessment models, site risk studies, and forensic analysis of the Fukushima accident.

How Has It Been Assessed?
MELCOR has been validated against numerous international standard problems, benchmarks, separate effects (e.g., VERCORS) and integral experiments (e.g., Phebus FPT), and reactor accidents (e.g., TMI-2, Fukushima).

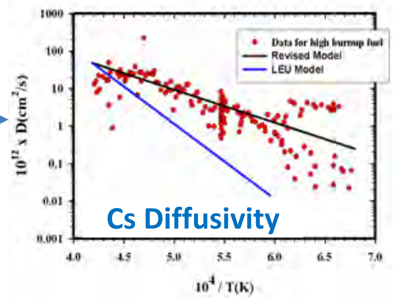
Source Term Development Process

Experimental Basis ← PIRT process

- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

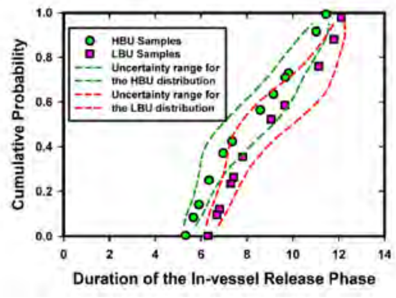
MELCOR

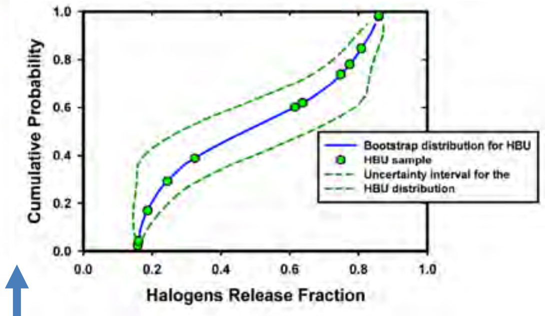
Cesium release from high burnup fuel
Comparison to results of VERCORS Test



Accident Analysis

Scenario # 1	Scenario # 2
……………….	……………….
Scenario # n-1	Scenario # n

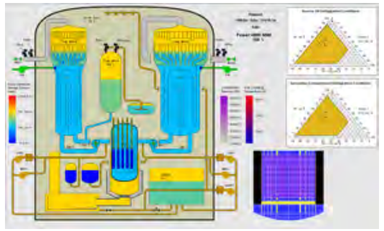


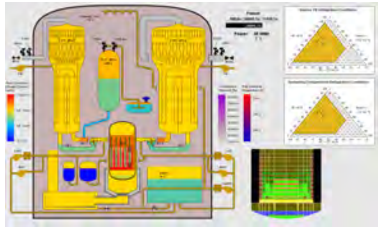


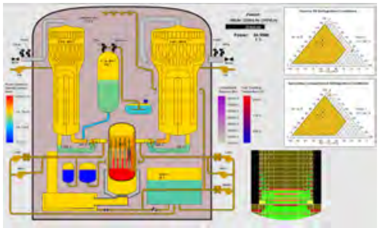
Synthesize timings and release fractions

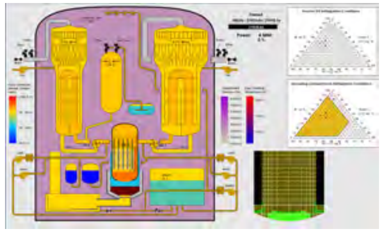
Design Basis Source Term

Fission Product class	Gap 0.5 hr	In-vessel 1.3 hr	Ex-vessel 2 hr	Late 10 hr
Noble Gases	5%	95%	~0	~0
Iodine, bromine	5%	35%	25%	1%
Cesium	5%	25%	35%	1%
Tellurium	~0	5%	25%	0.5%
Ba, Sr	~0	2%	10%	~0
Ru, Mo, Pd, etc.	~0	0.25%	0.25%	~0
Lanthanides	~0	0.02%	0.5%	~0
Cerium group	~0	0.05%	0.5%	~0









SCALE/MELCOR/MACCS

U.S.NRC
Oak Ridge National Laboratory
Sandia National Laboratories

SCALE — Neutronics

- Criticality
- Shielding
- Radionuclide inventory
- Burnup credit
- Decay heat

MELCOR — Integrated Severe Accident Progression

- Hydrodynamics for range of working fluids
- Accident response of plant structures, systems and components
- Fission product transport

MACCS — Radiological Consequences

- Near- and far-field atmospheric transport and deposition
- Assessment of health and economic impacts

Nuclear Reactor System Applications

Safety/Risk Assessment

- Technology-neutral
  - Experimental
  - Naval
  - Advanced LWRs
  - Advanced Non-LWRs
- Accident forensics (Fukushima, TMI)
- Probabilistic risk assessment

Regulatory

- License amendments
- Risk-informed regulation
- Design certification (e.g., NuScale)
- Vulnerability studies
- Emergency preparedness
- Emergency Planning Zone Analysis

Design/Operational Support

- Design analysis scoping calculations
- Training simulators

Non-Reactor Applications

Fusion

- Neutron beam injectors
- Li loop LOFA transient analysis
- ITER cryostat modeling
- He-cooled pebble test blanket (H^3)

Spent Fuel

- Risk studies
- Multi-unit accidents
- Dry storage
- Spent fuel transport/package applications

Facility Safety

- Leak path factor calculations
- DOE safety toolbox codes
- DOE nuclear facilities (Pantex, Hanford, Los Alamos, Savannah River Site)

MELCOR Attributes
Foundations of MELCOR Development

Fully integrated, engineering-level code

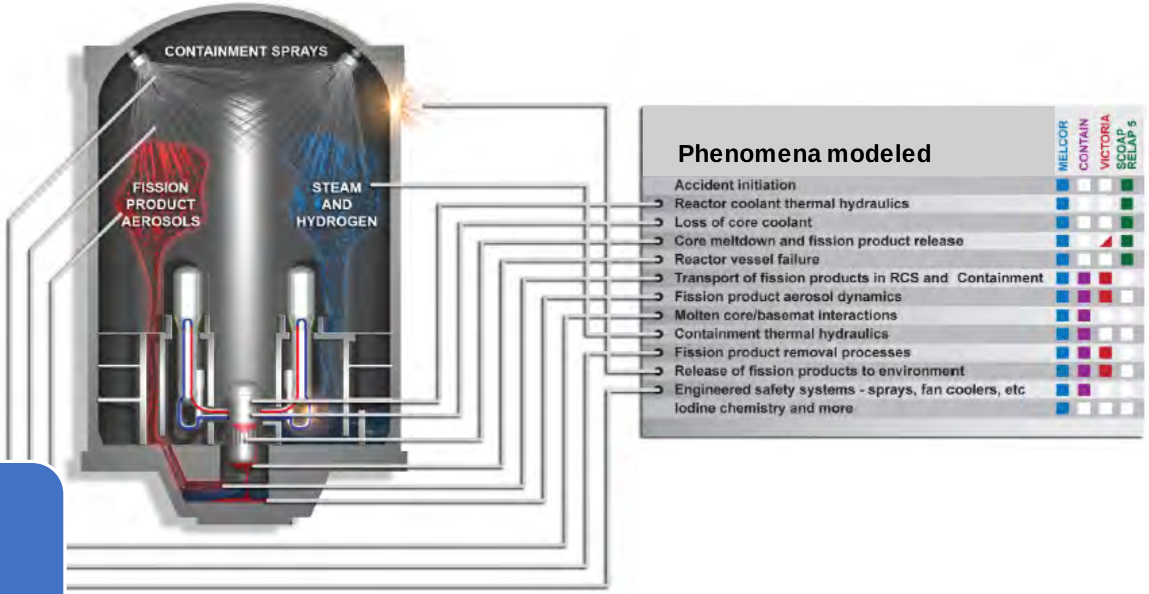
- Thermal-hydraulic response of reactor coolant system, reactor cavity, rector enclosures, and auxiliary buildings
- Core heat-up, degradation and relocation
- Core-concrete interaction
- Flammable gas production, transport and combustion
- Fission product release and transport behavior

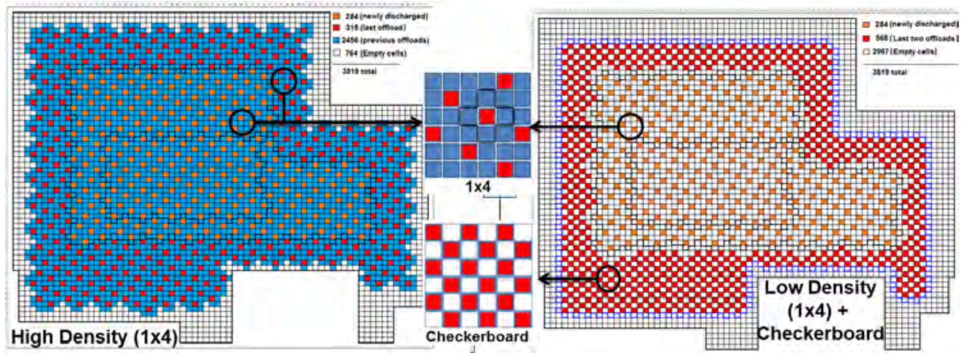
Level of physics modeling consistent with

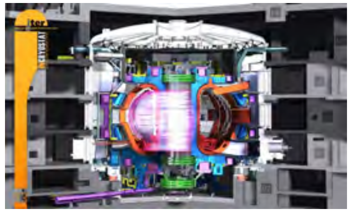
- State-of-knowledge
- Necessity to capture global plant response
- Reduced-order and correlation-based modeling often most valuable to link plant physical conditions to evolution of severe accident and fission product release/transport

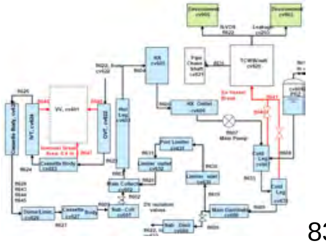
Traditional application

- Models constructed by user from basic components (control volumes, flow paths and heat structures)
- Demonstrated adaptability to new reactor designs – HPR, HTGR, SMR, MSR, ATR, Naval Reactors, VVER, SFP,…









MELCOR Attributes

MELCOR Pedigree

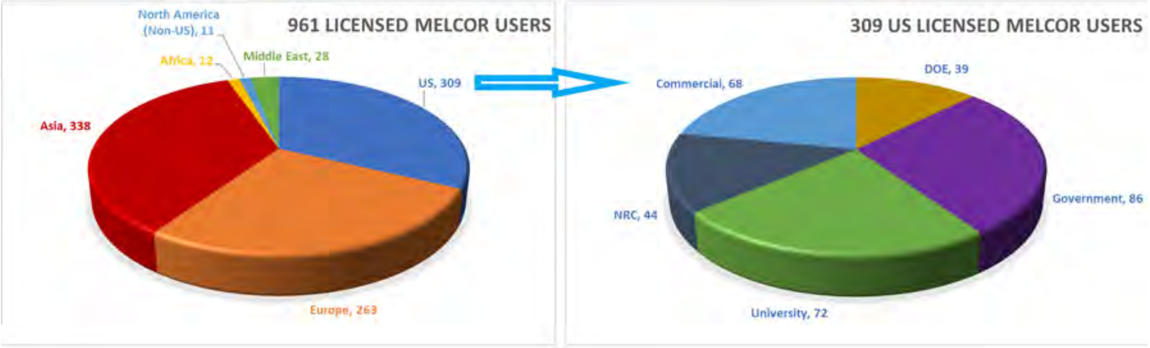
Validated physical models

- 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

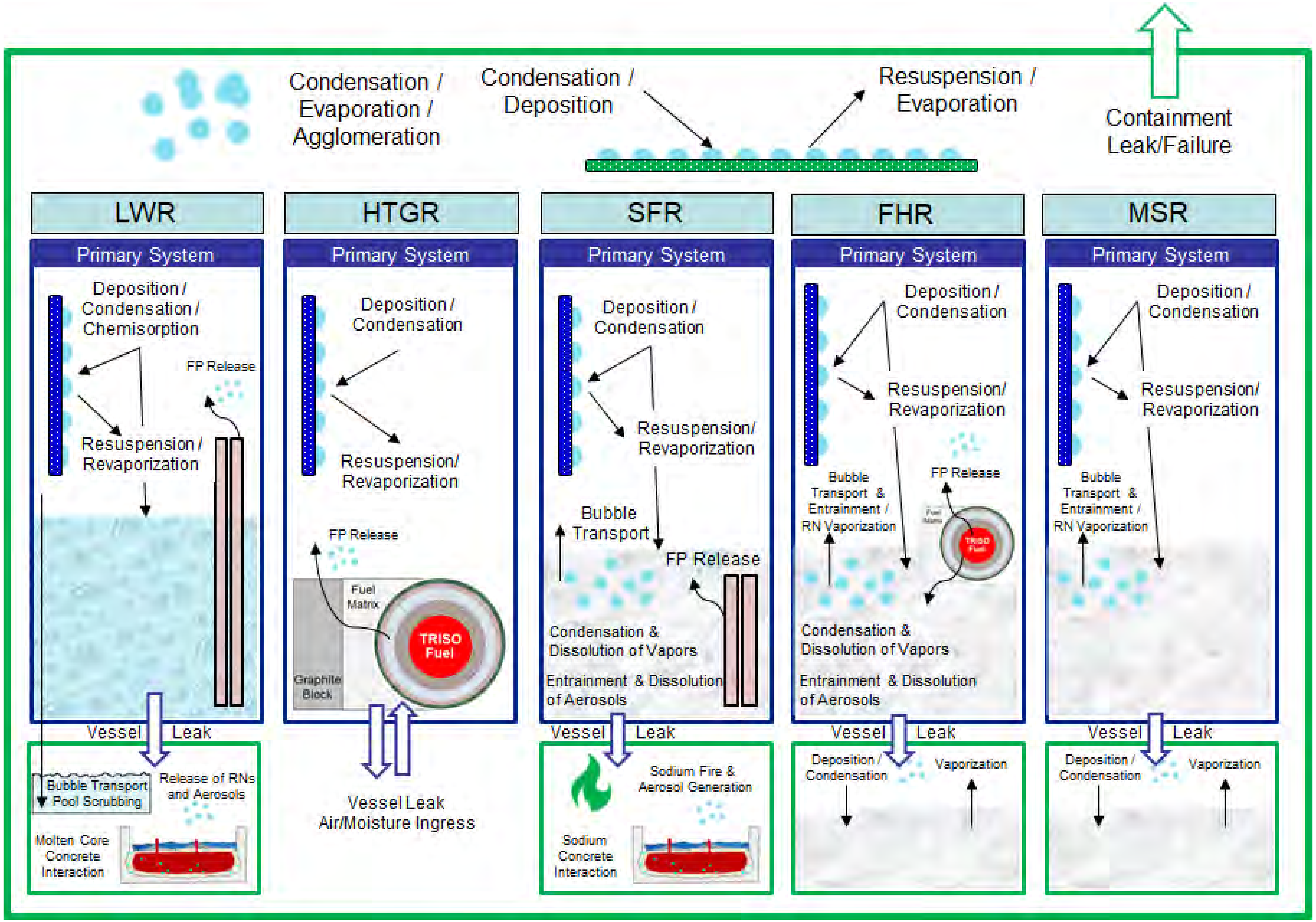
Cooperative Severe Accident Research Program (CSARP) is an NRC-sponsored international, collaborative community supporting the validation of MELCOR

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



Common Phenomenology



MELCOR Modeling Approach

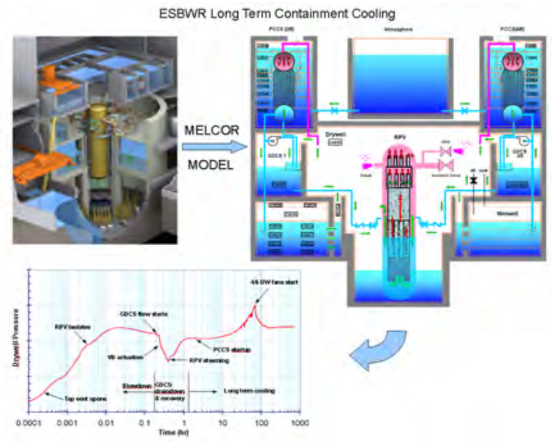
Modeling is mechanistic consistent with level of knowledge of phenomena supported by experiments

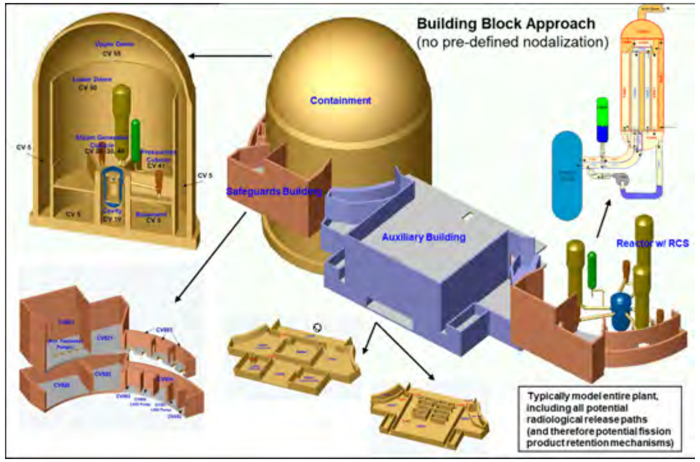
Parametric models enable uncertainties to be characterized

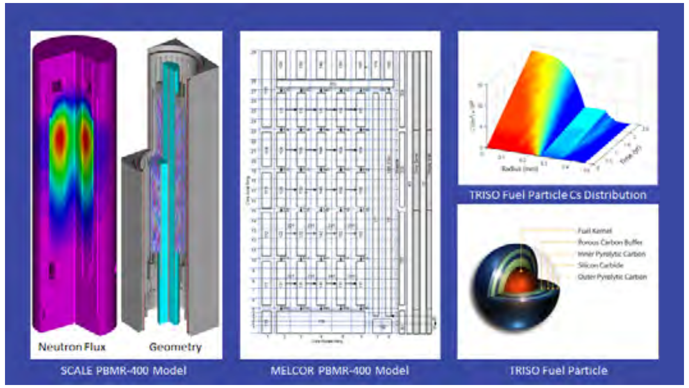
- Majority of modeling parameters can be varied
- Properties of materials, correlation coefficients, numerical controls/tolerances, etc.

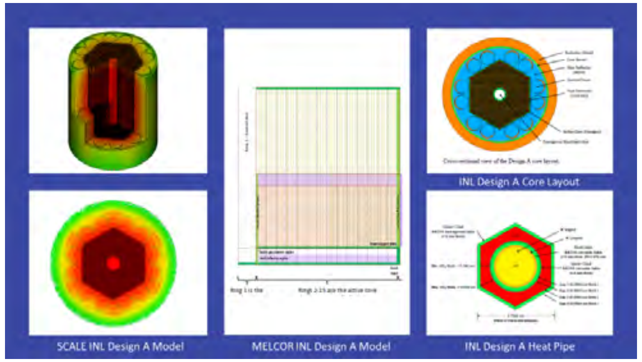
Code models are general and flexible

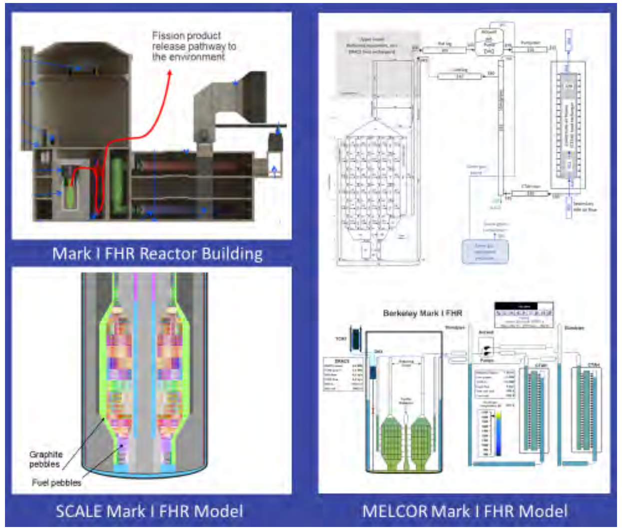
- Relatively easy to model novel designs
- All-purpose thermal hydraulic and aerosol transport code







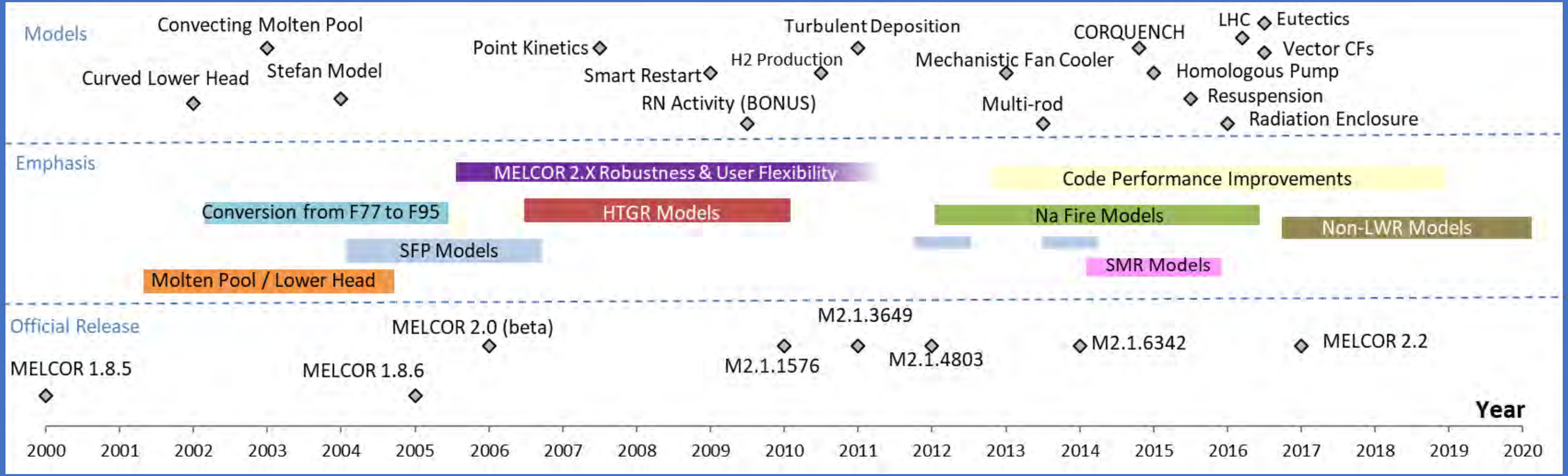




MELCOR State-of-the-Art

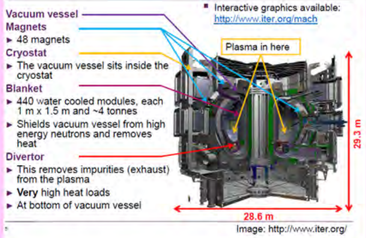
U.S.NRC
OAK RIDGE National Laboratory
Sandia National Laboratories

MELCOR Code Development



M2x Official Code Releases

Version	Date
2.2.18180	December 2020
2.2.14959	October 2019
2.2.11932	November 2018
2.2.9541	February 2017
2.1.6342	October 2014
2.1.4803	September 2012
2.1.3649	November 2011
2.1.3096	August 2011
2.1.YT	August 2008
2.0 (beta)	Sept 2006



Fusion

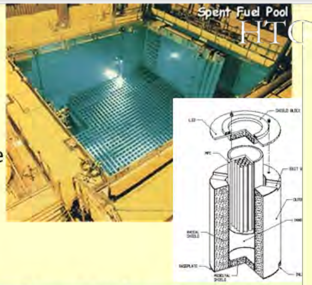
- Neutron Beam Injectors (LOVA)
- Li Loop LOFA transient analysis
- ITER Cryostat modeling
- Helium Lithium
- Helium Cooled Pebble Bed Test Blanket (Tritium Breeding)

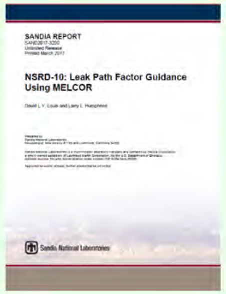
Spent Fuel

Spent fuel pool risk studies

Multi-unit accidents (large area destruction)

Dry Storage



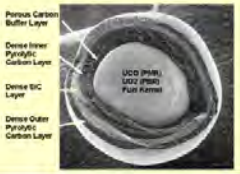


Non-Nuclear Facilities

- Leak Path Factor Calculations (LPF)
  - Release of hazardous materials from facilities, buildings, confined spaces
- DOE Safety Toolbox code
- DOE nuclear facility users
  - Pantex
  - Hanford
  - Los Alamos
  - Savannah River Site

HTGR Reactors

- Helium Properties
- Accelerated steady-state initialization
- Two-sided reflector (RF) component
- Modified Fuel components (PMR/PBR)
- Point kinetics
- Fission product diffusion, transport, and release
- TRISO fuel failure



Sodium Reactors

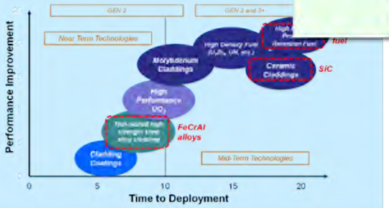
Sodium Properties
- Sodium Equation of State
- Sodium Thermo-mechanical properties

Containment Modeling
- Sodium pool fire model
- Sodium spray fire model
- Atmospheric chemistry model
- Sodium-concrete interaction

Molten Salt Reactors

- Properties for LiF-BeF2 have been added
  - Equation of State
  - Thermal-mechanical properties

Accident Tolerant Fuels



MELCOR Software Quality Assurance – Best Practices

U.S.NRC
OAK RIDGE National Laboratory
Sandia National Laboratories

MELCOR SQA Standards

SNL Corporate procedure IM100.3.5
CMMI-4+
NRC NUREG/BR-0167

Emphasis is on Automation

Affordable solutions
Consistent solutions

MELCOR Wiki

- Archiving information
- Sharing resources (policies, conventions, information, progress) among the development team.

Code Configuration Management (CM)

- ‘Subversion’
- TortoiseSVN
- VisualSVN integrates with Visual Studio (IDE)

Reviews

- Code Reviews: Code Collaborator
- Internal SQA reviews

Continuous builds & testing

- DEF application used to launch multiple jobs and collect results
- Regression test report
- More thorough testing for code release
- Target bug fixes and new models for testing

Bug tracking and reporting

- Bugzilla online

Code Validation

- Assessment calculations
- Code cross walks for complex phenomena where data does not exist.

Documentation

- Available on ‘Subversion’ repository with links from wiki
- Latest PDF with bookmarks automatically generated from word documents under Subversion control
- Links on MELCOR wiki

Project Management

- Jira for tracking progress/issues
- Can be viewable externally by stakeholders

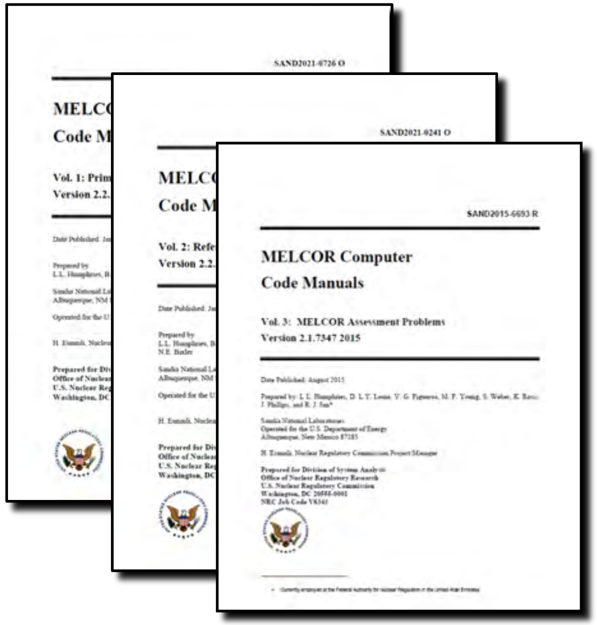
Sharing of information with users

- External web page
- MELCOR workshops
- MELCOR User Groups (EMUG & AMUG)

Case	BUR	CAV	CF	COR	CVH	DCH	FCL	FDI	FL	HS	NCG	PAR	RN	SPR
M-8-1 NoMix			X		X				X	X	X			
M-8-1 SYM			X		X				X	X	X			
Lace7			X		X	X			X	X	X		X	
Lace8			X		X	X			X	X	X		X	
Vanam-M3			X		X				X	X	X		X	
Molten Salt			X	X	X				X	X	X			
PHEBUS-B9			X	X	X				X	X	X			
FPT1			X	X	X	X			X	X	X		X	
LOFT			X	X	X	X			X	X	X			
Test Inew	X	X	X	X	X	X	X	X	X	X	X	X	X	X
SURRY (LBLOCA)	X	X	X	X	X	X	X	X	X	X	X		X	X
Zion (SBO)		X	X	X	X	X	X	X	X	X	X	X	X	X
PeachBottom (SBO)	X	X	X	X	X	X			X	X	X		X	X
Grand Gulf (SBO)	X	X	X	X	X	X		X	X	X	X		X	

Table 1-1: Physics Package Coverage

MELCOR Verification & Validation Basis

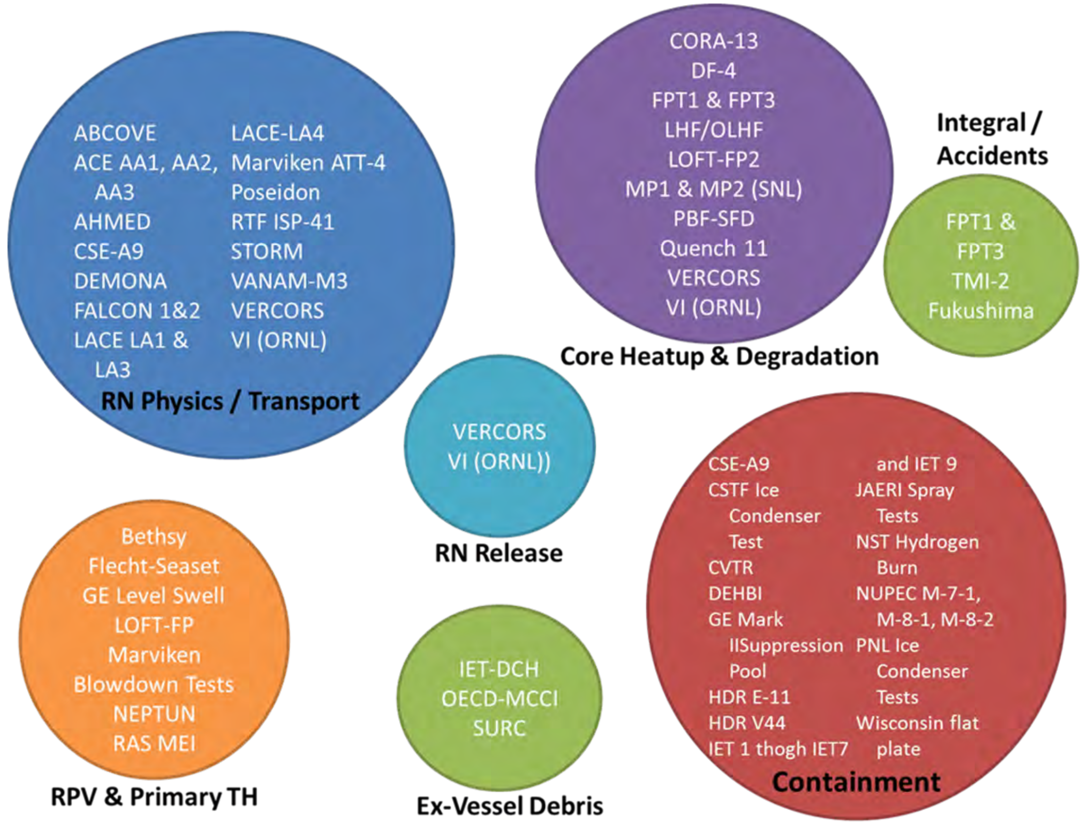


Volume 1: Primer & User Guide
Volume 2: Reference Manual
Volume 3: MELCOR Assessment Problems
[SAND2015-6693 R]

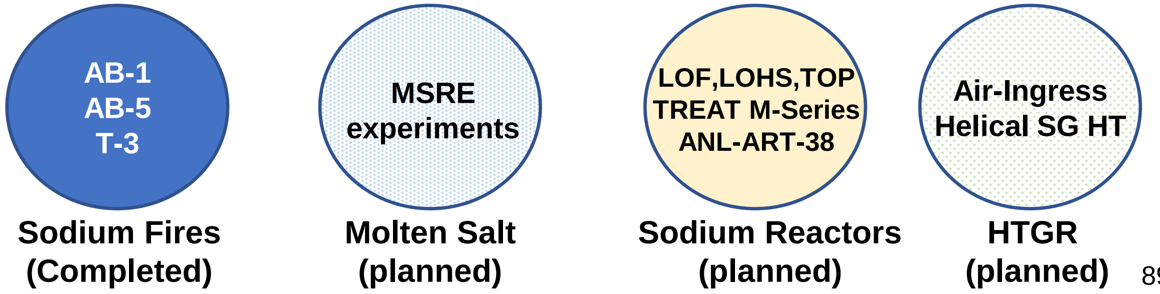
Analytical Problems

Saturated Liquid Depressurization
Adiabatic Expansion of Hydrogen
Transient Heat Flow in a Semi-Infinite Heat Slab
Cooling of Heat Structures in a Fluid
Radial Heat Conduction in Annular Structures
Establishment of Flow

LWR & non-LWR applications



Specific to non-LWR application



Sample Validation Cases

TRISO Diffusion Release

IAEA CRP-6 Benchmark
Fractional Release

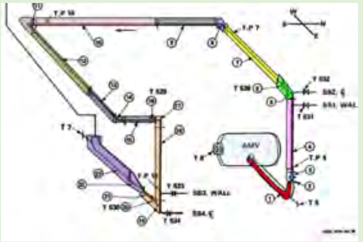
Case	1a	1b	2a	2b	3a	3b
US/INL	0.467	1.0	0.026	0.996	1.32E-4	0.208
US/GA	0.453	0.97	0.006	0.968	7.33E-3	1.00
US/SNL	**0.465**	**1.0**	**0.026**	**0.995**	**1.00E-4**	**0.208**
US/NRC	0.463	1.0	0.026	0.989	1.25E-4	0.207
France	0.472	1.0	0.028	0.995	6.59E-5	0.207
Korea	0.473	1.0	0.029	0.995	4.72E-4	0.210
Germany	0.456	1.0	0.026	0.991	1.15E-3	0.218

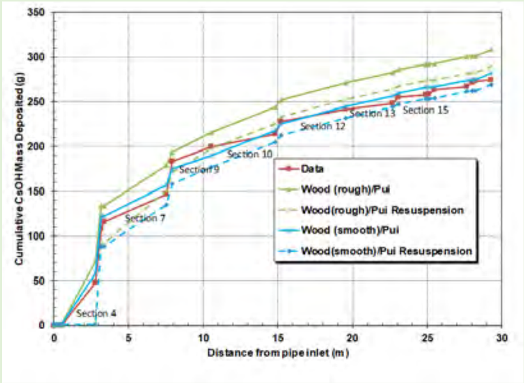
(1a): Bare kernel (1200 ºC for 200 hours)
(1b): Bare kernel (1600 ºC for 200 hours)
(2a): kernel+buffer+iPyC (1200 ºC for 200 hours)
(2b): kernel+buffer+iPyC (1600 ºC for 200 hours)
(3a): Intact (1600 ºC for 200 hours)
(3b): Intact (1800 ºC for 200 hours)

A sensitivity study to examine fission product release from a fuel particle starting with a bare kernel and ending with an irradiated TRISO particle;

Turbulent Deposition

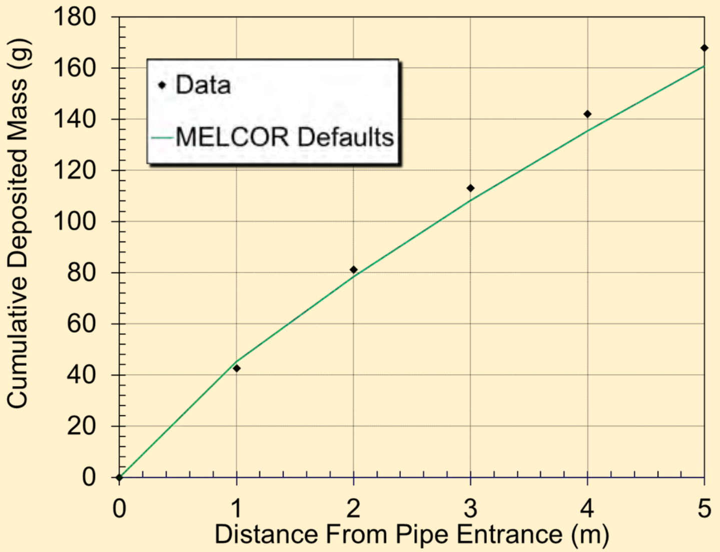
LACE LA1 and LA3 tests experimentally examined the transport and retention of aerosols through pipes with high speed flow





Resuspension

STORM (Simplified Test of Resuspension Mechanism) test facility

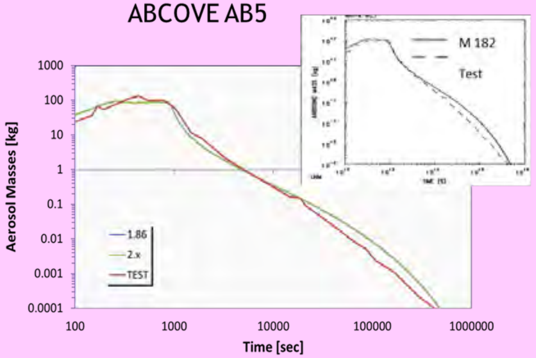


Aerosol Physics

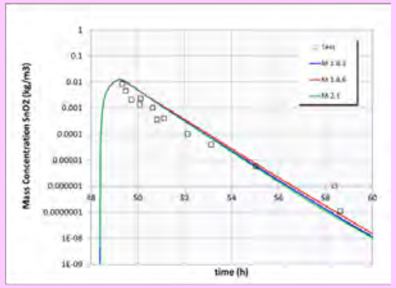
- Agglomeration
- Deposition
- Condensation and Evaporation at surfaces

Validation Cases
- Simple geometry: AHMED, ABCOVE (AB5 & AB6), LACE(LA4),
- Multi-compartment geometry: VANAM (M3), DEMONA(B3)
- Deposition: STORM, LACE(LA1, LA3)

ABCOVE AB5

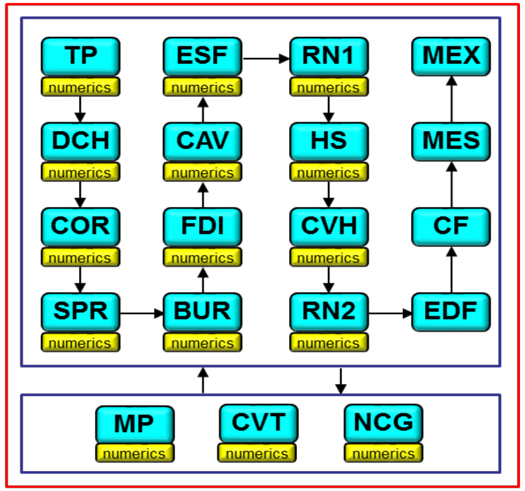


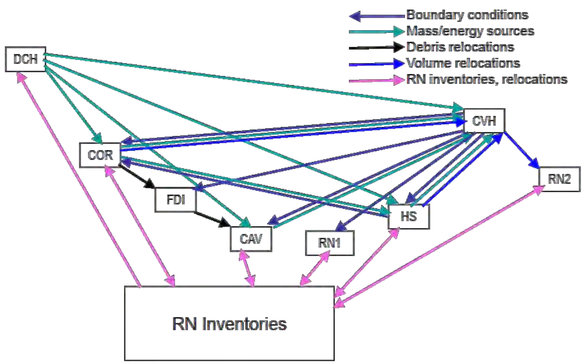
DEMONA-B3



MELCOR Modernization

- Generalized numerical solution engine
- Hydrodynamics
- In-vessel damage progression
- Ex-vessel damage progression
- Fission product release and transport





TP = Transfer Process
DCH = Decay Heat
COR = Core
SPR = Containment Spray
BUR = Gas Combustion
FDI = Fuel Dispersal Interaction
CAV = Cavity (MCCI)
ESF = Engineered Safety Features
MP = Material Properties

RN = Radionuclide
HS = Heat Structure
CVH = CV Hydrodynamics
EDF = External Data File
CF = Control Function
MES = Special Messages
MEX = Executive
CVT = CV Thermodynamics
NCG = Non Condensible Gas

Separate **Physics** & **Numerics**