SAND2020-8493PE

Advanced-Reactor Source Term – Pilot Studies

Advanced Reactor Stakeholder Meeting

August 20, 2020

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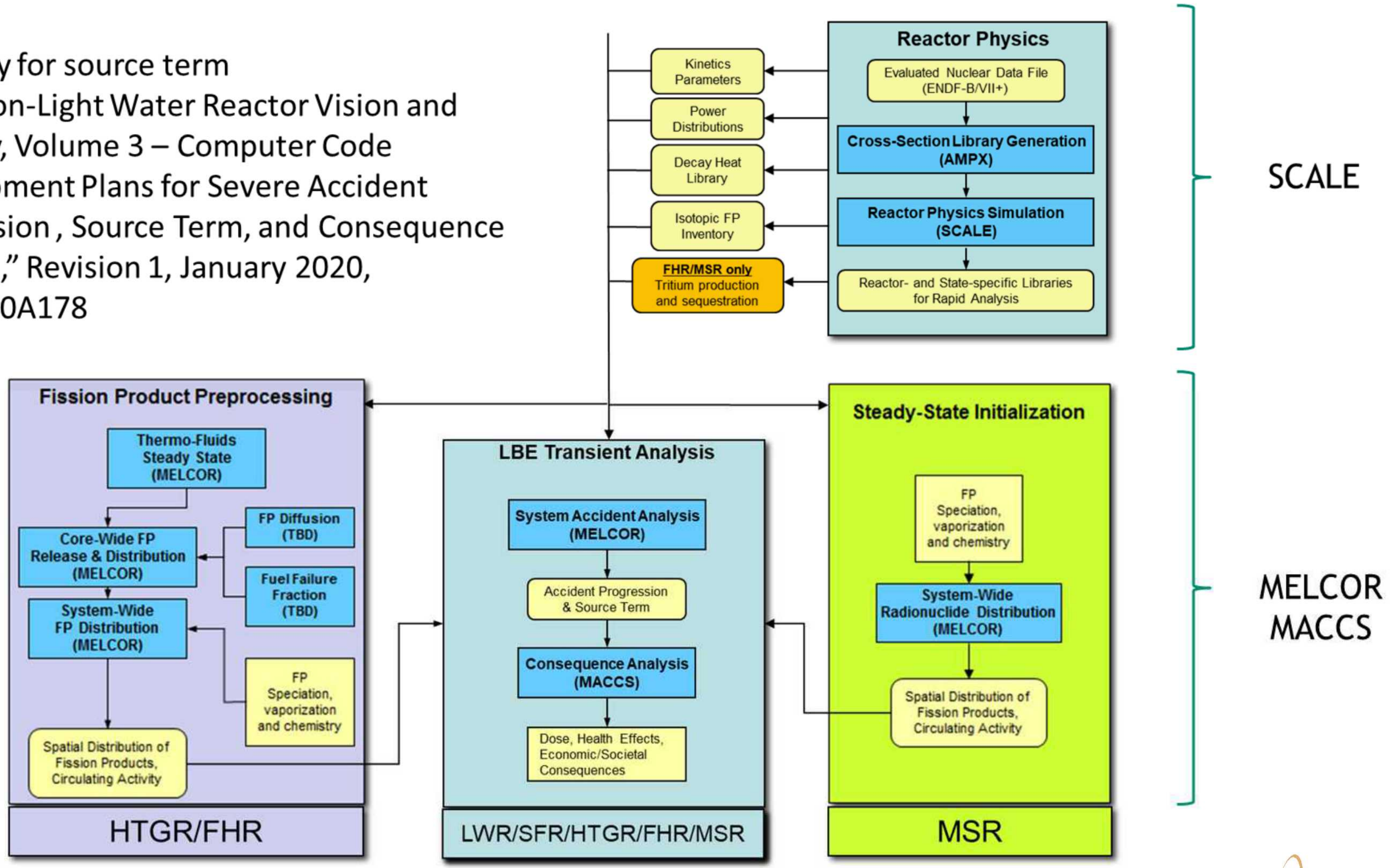
Non-LWR Evaluation Model

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



Project Objectives

Develop an understanding of non-LWR beyond-design-basis-accident behavior

- Provide insights for regulatory guidance
- Facilitate dialogue on the staff's approach to assessing source term

Demonstrate application of MELCOR and SCALE

- Develop publicly available input models - available upon request
- Code distribution handled separately

Project Stages

1. Select design
2. Develop input deck
3. Select scenarios
4. **Perform calculations and refine input deck**
   - Full-plant decks have been developed for heat pipe and gas-cooled reactors
   - Salt-cooled reactor input deck in preparation
   - Results shown here are preliminary to illustrate approach
5. Public workshop

INL Design A Heat Pipe Reactor


INL/EXT-17-43212
Revision 1

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

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Robert C. O'Brien
Axel M. Dion
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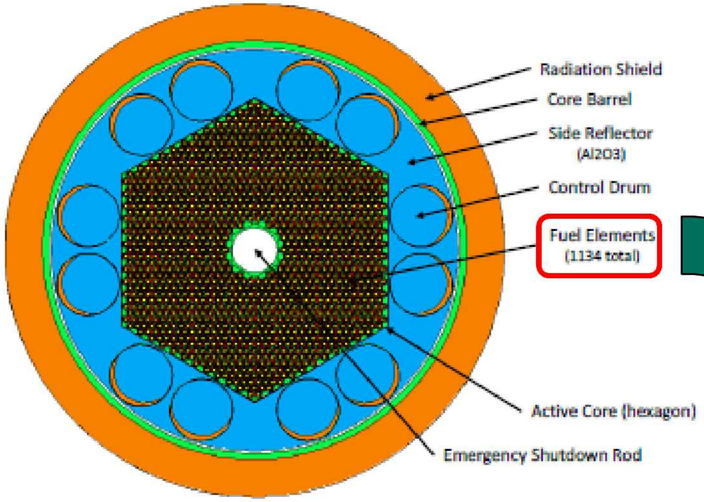
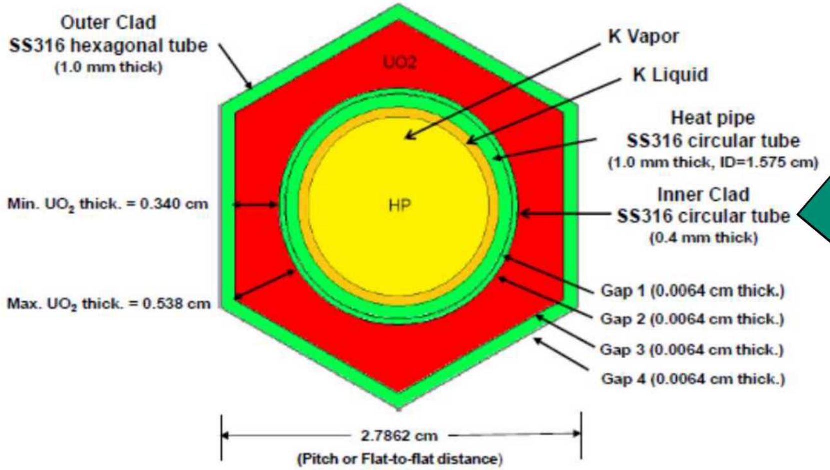
May 2018

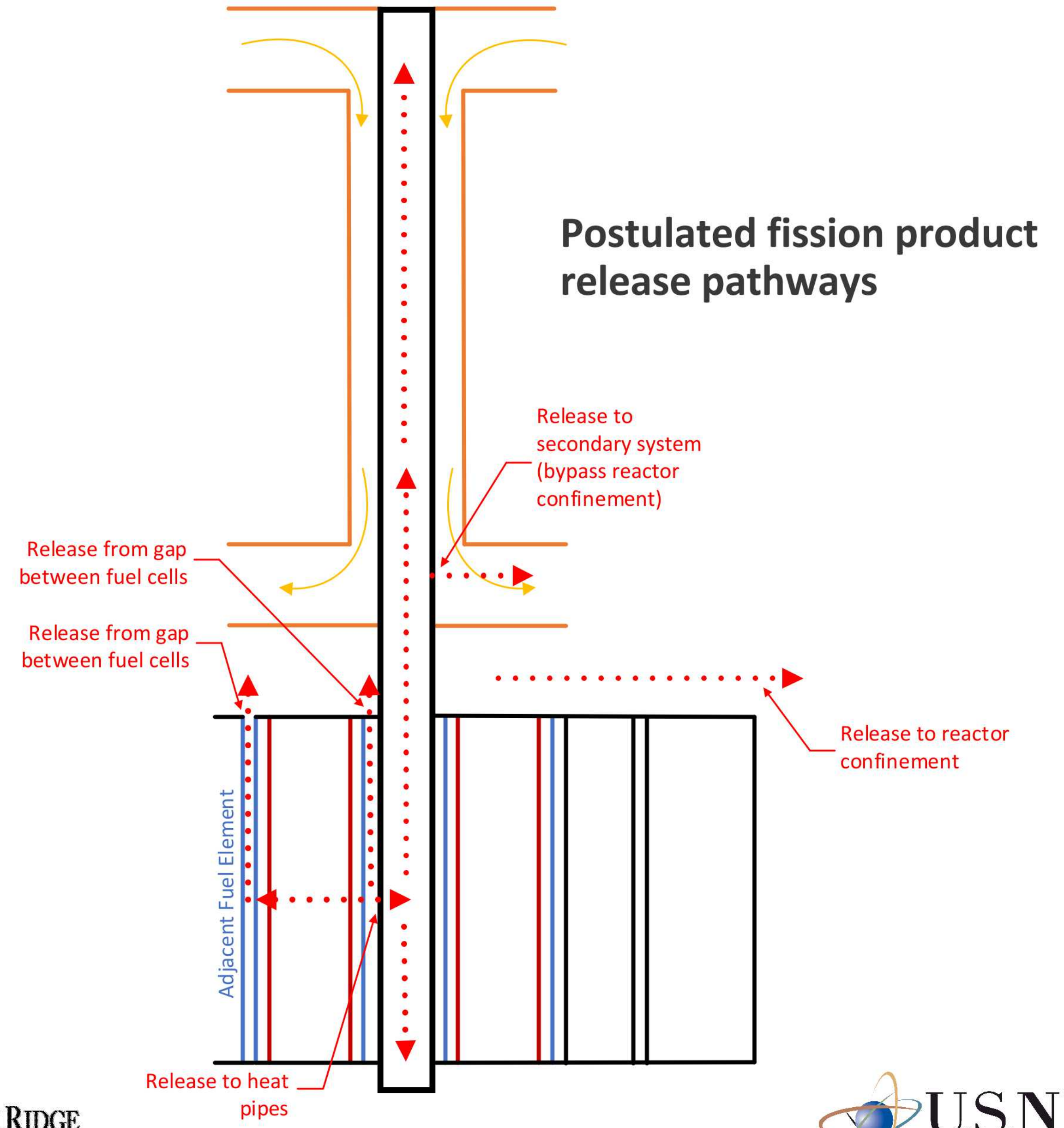
Idaho National Laboratory

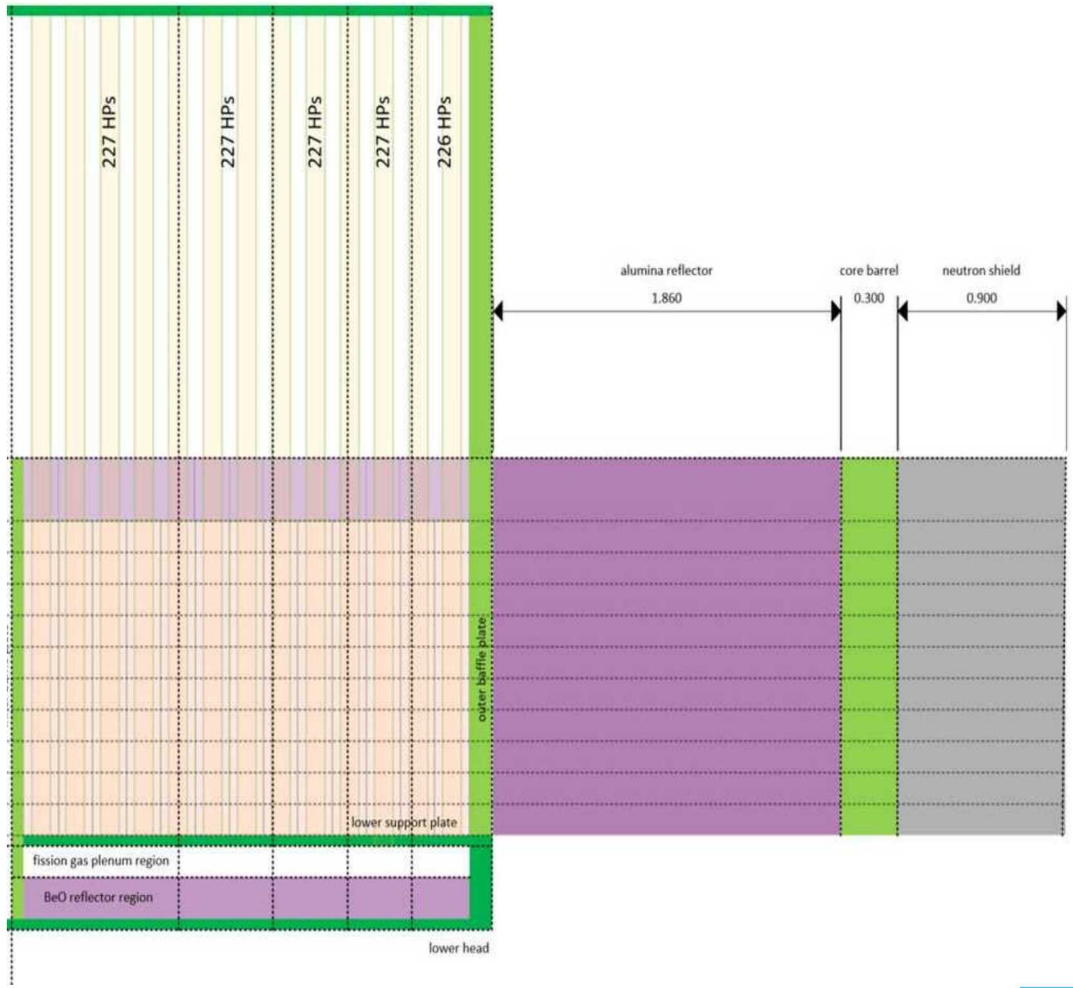


Cross-sectional view of the Design A core layout.

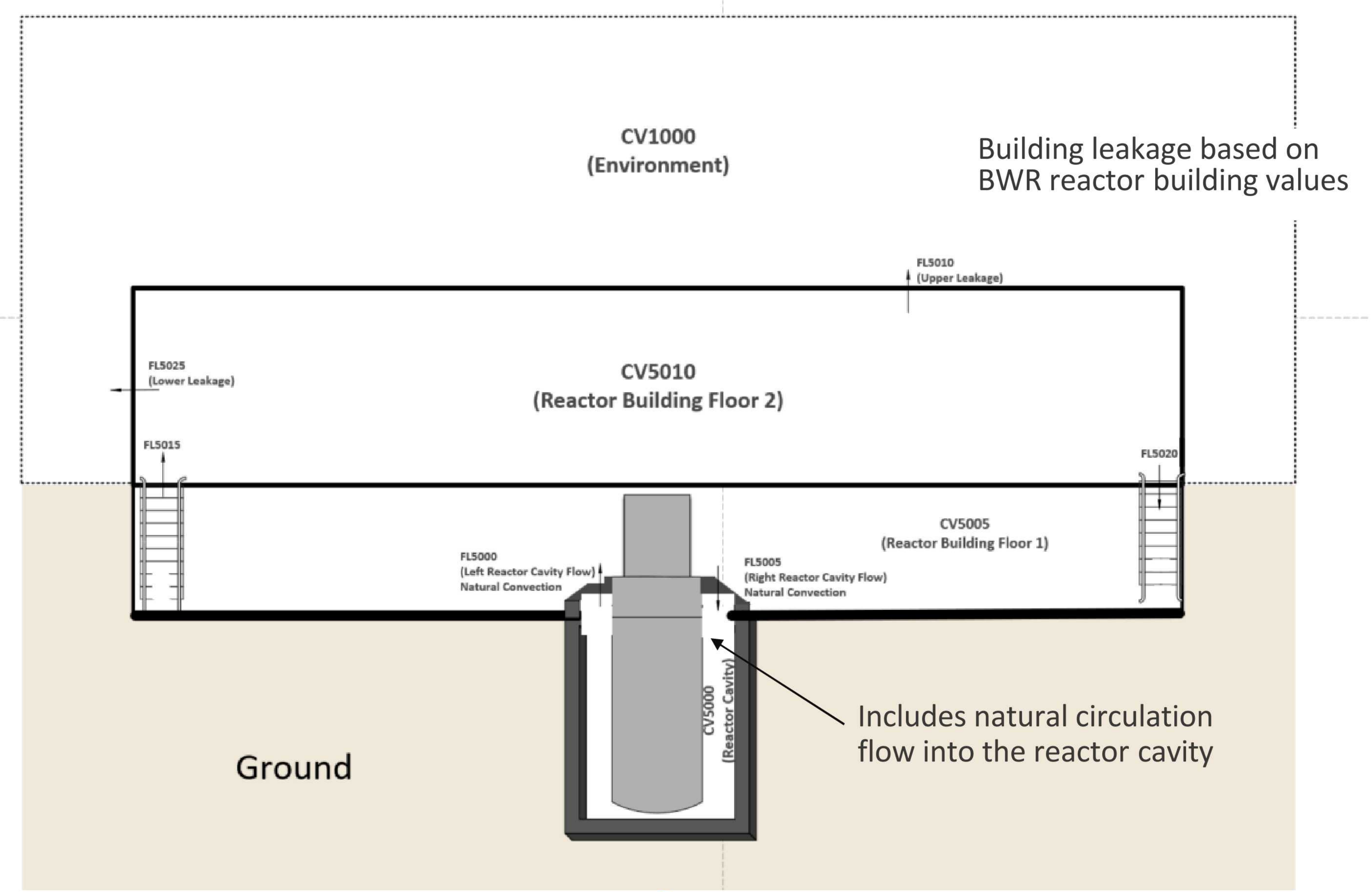


INL Design A – Reactor vessel and core nodalization





INL Design A – Reactor building nodalization

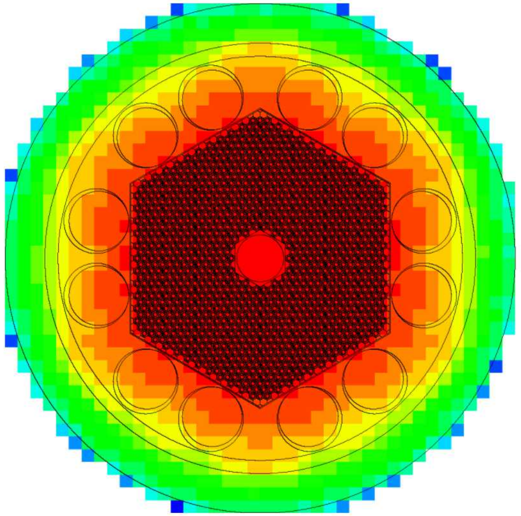


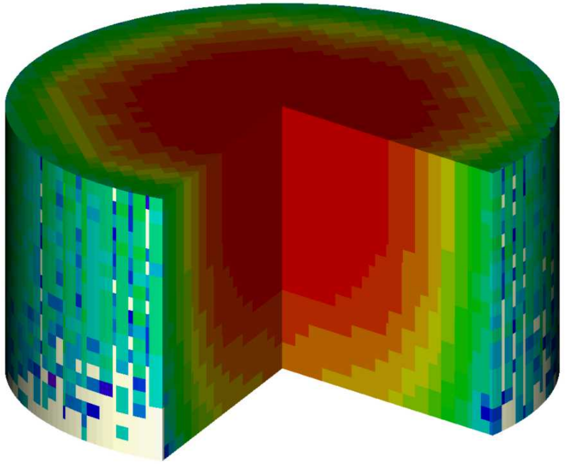
Building leakage based on BWR reactor building values

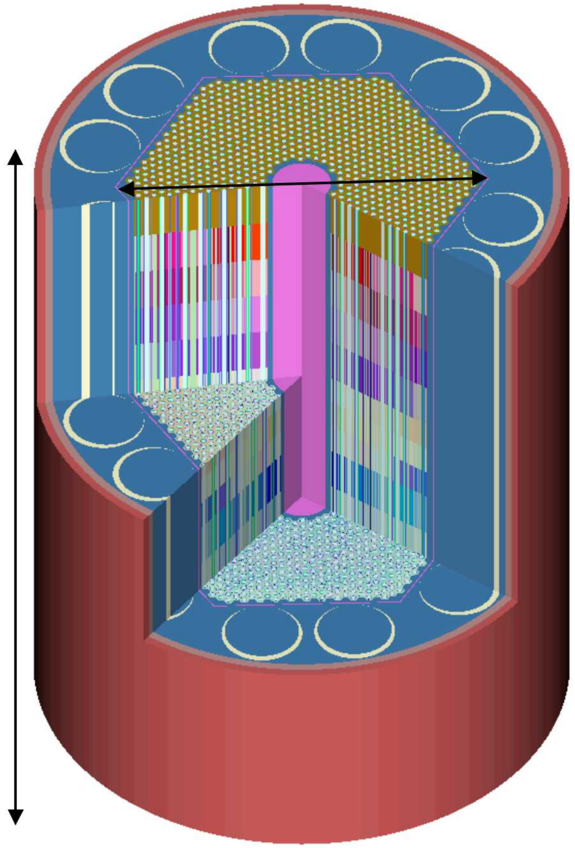
Includes natural circulation flow into the reactor cavity

INL Design A SCALE model

- Design features
  - 1134 annular hexagonal UO_2 fuel elements (19.75% ^{235}U)
  - Fast neutron spectrum
- Modeling strategy
  - Flux was evaluated assuming a fixed control drum configuration
  - Isotopic inventory evaluated at full power over core life
- Radionuclide inventory and decay heat data provided for MELCOR model



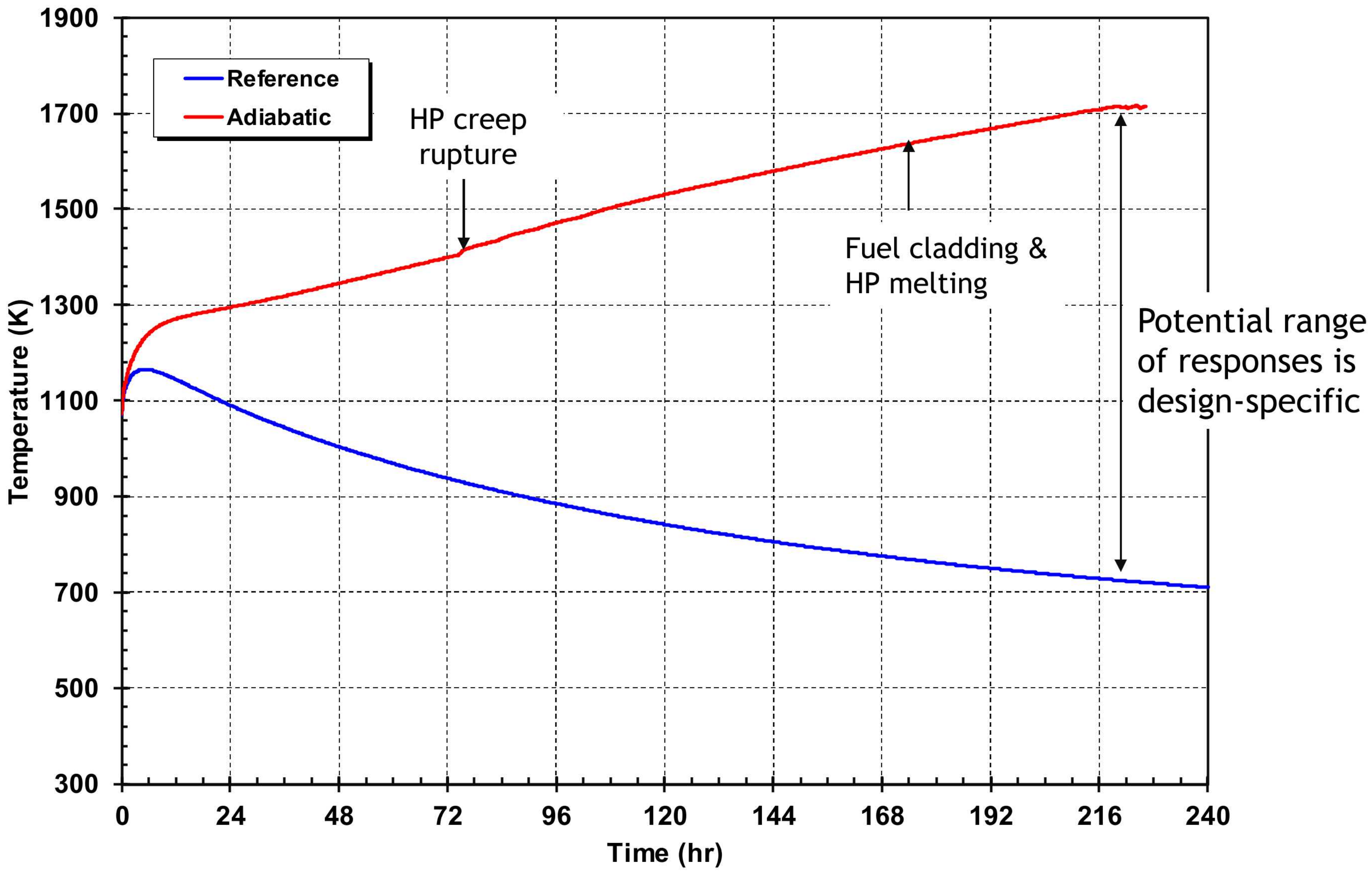




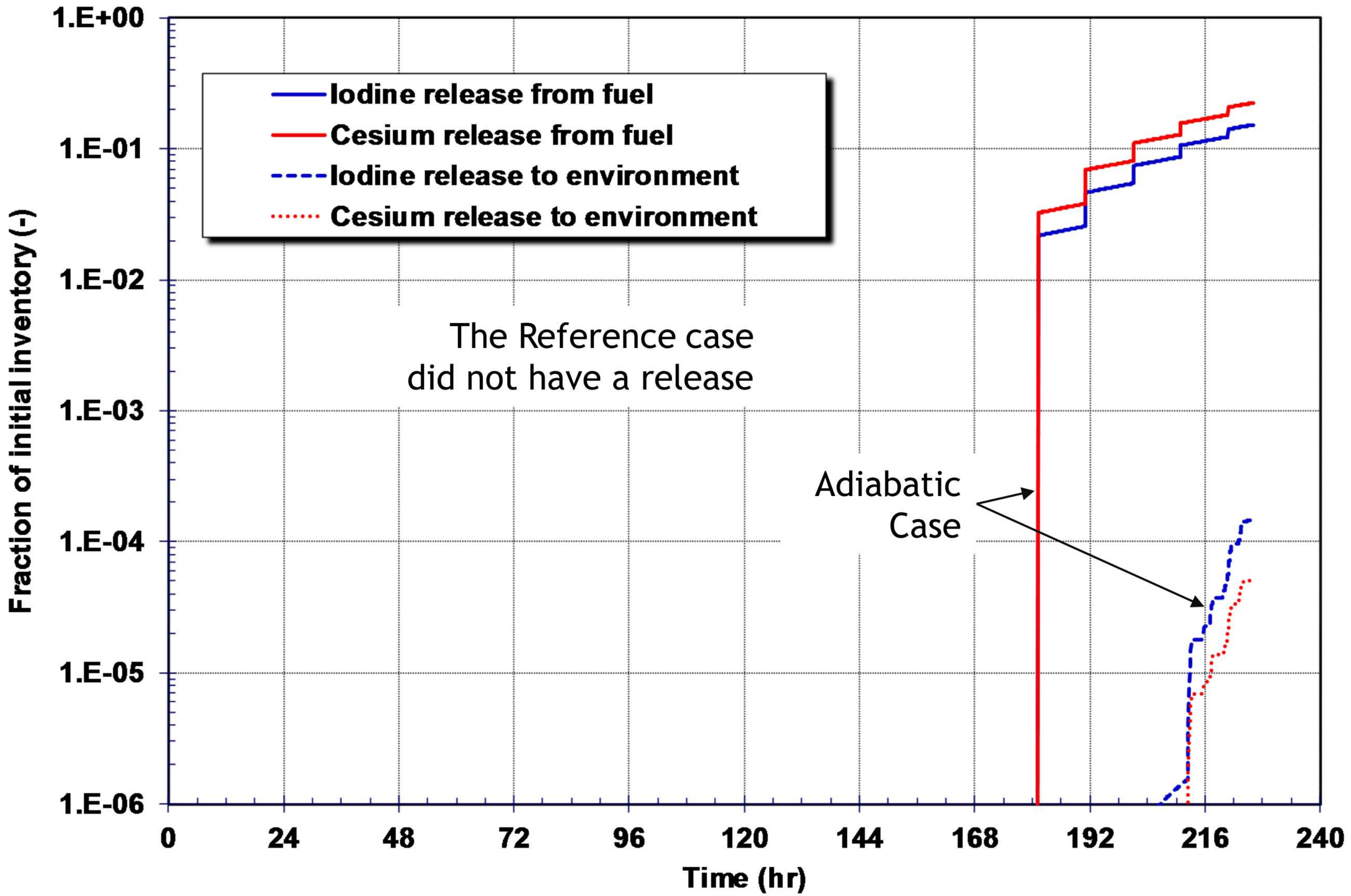
INL Design A – Demo calculations

- Reference case
  - Initiator trips secondary heat removal
  - Control rod insertion
  - Thermal radiation from the reactor vessel
  - Natural circulation flow through the reactor cavity
- Adiabatic case
  - No convective or radiative heat transfer from the vessel

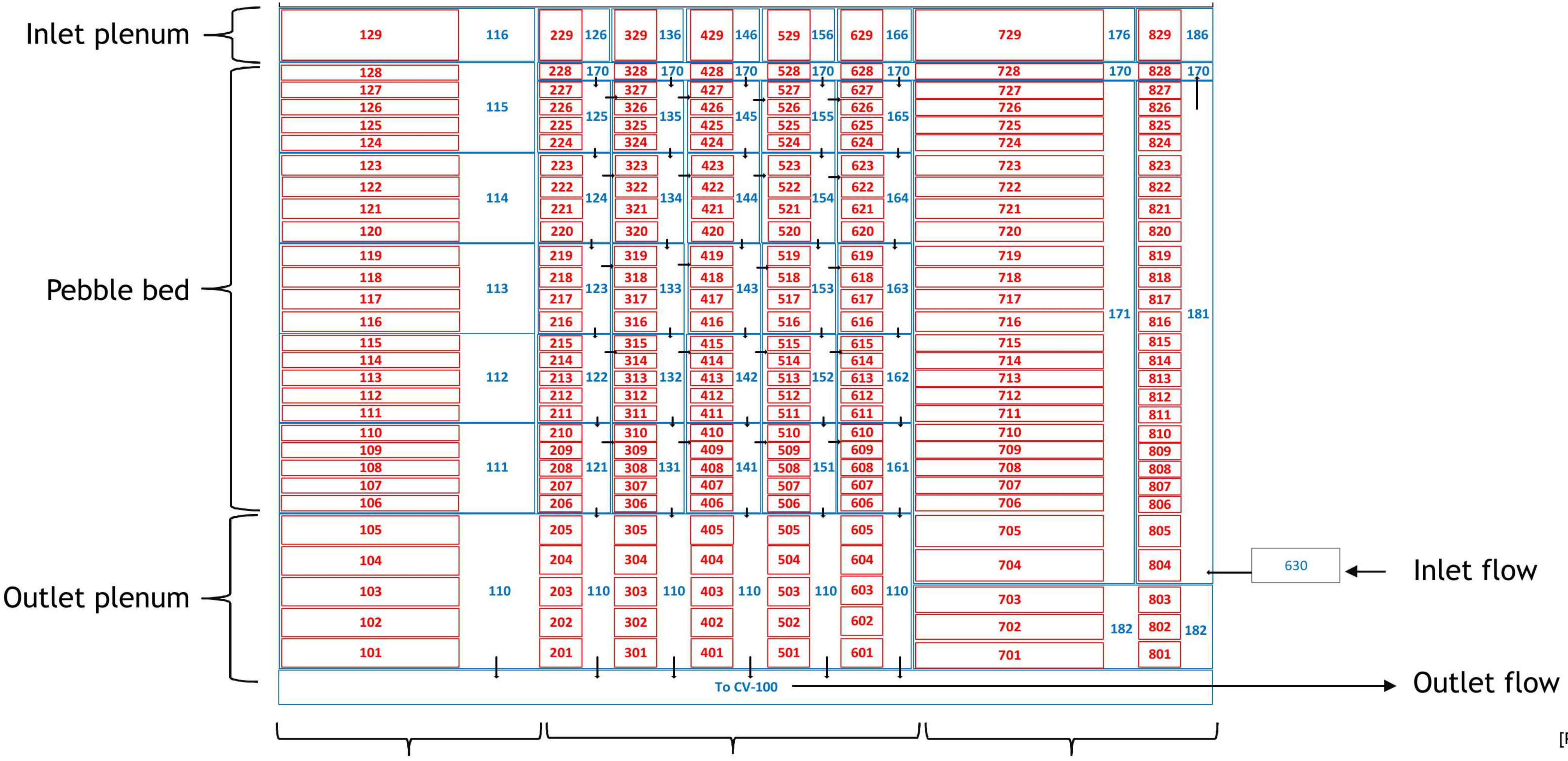
INL Design A – Peak fuel temperatures



INL Design A – Iodine and cesium release



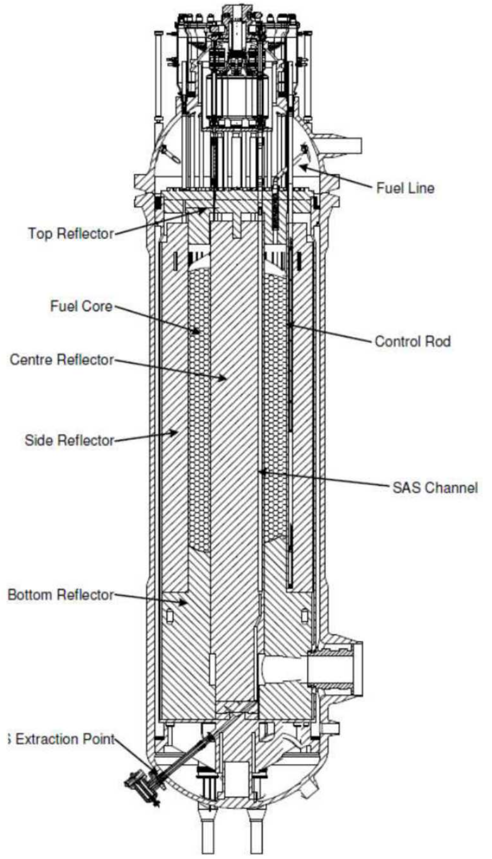
PBMR-400 reactor and core



Ring 1 is the graphite inner reflector

Pebble bed is in Rings 2-6

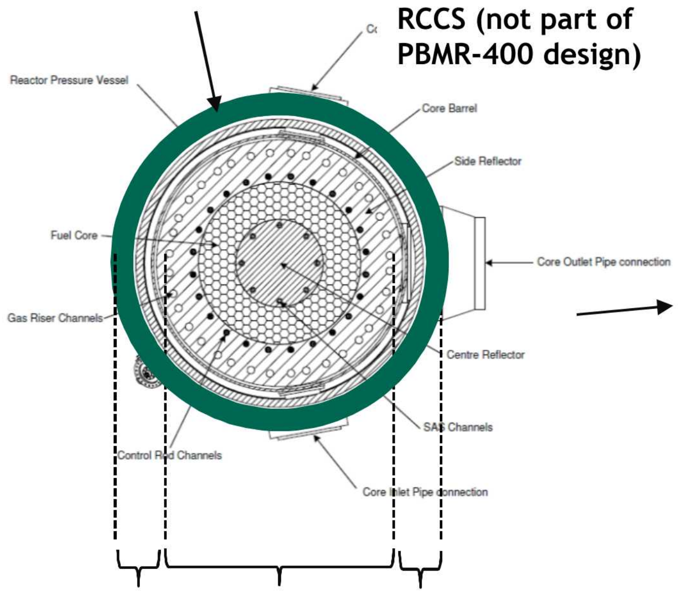
Ring 7 is the graphite reflector
Ring 8 is inlet flow risers



[P.J. Venter, M.N. Mitchell, F. Fortier, PBMR reactor design and development, in: Proceedings from the 18th International Conference on Structural Mechanics in Reactor Technology (SMiRT 18), Beijing, China, Aug. 2005]

PBMR-400 vessel and reactor building

RCCS (not part of PBMR-400 design)

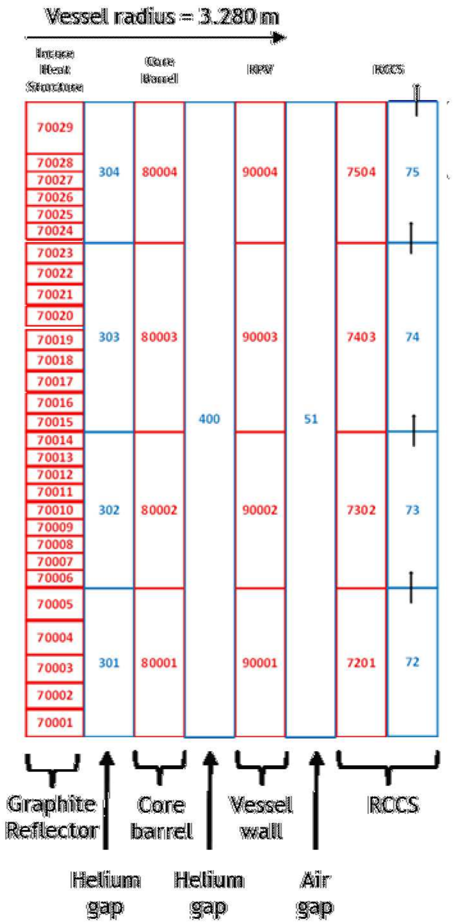


Core barrel
Vessel wall
RCCS

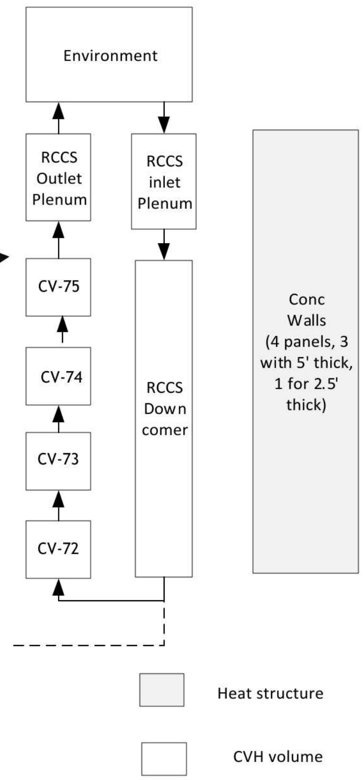
Rings 1- 8
(previous slide)

Core barrel
Vessel wall
RCCS

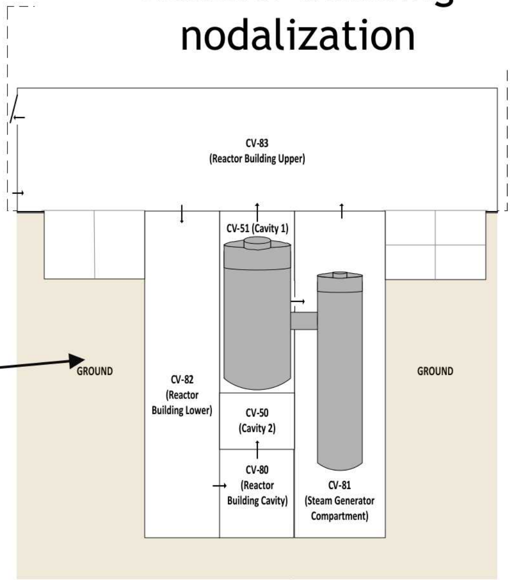
[Ducknor, Nuclear Engineering and Technology, 49, 360-372, 2017]

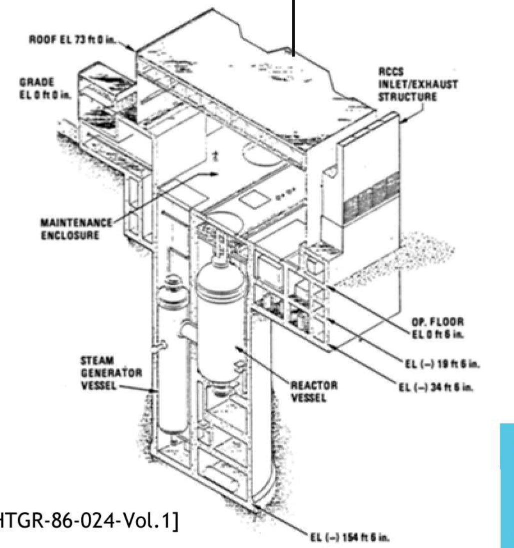


RCCS adapted from the from MHTGR



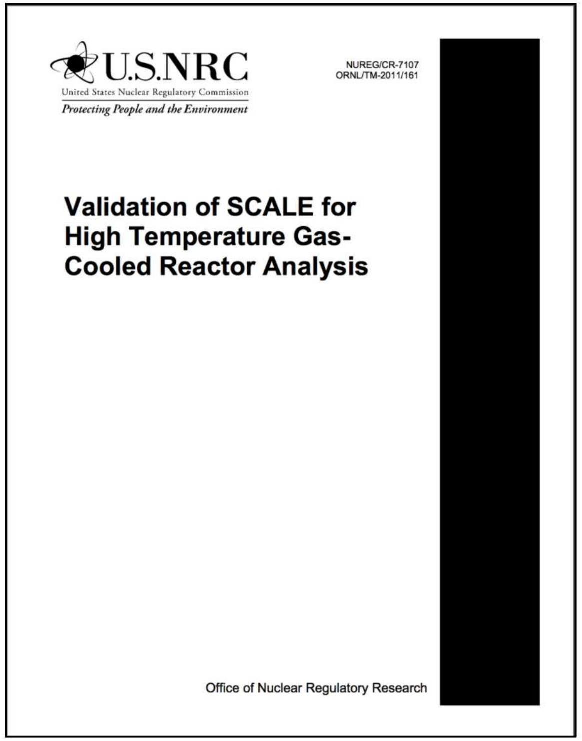
Reactor building nodalization





[DOE-HTGR-86-024-Vol.1]

PBMR-400 SCALE model

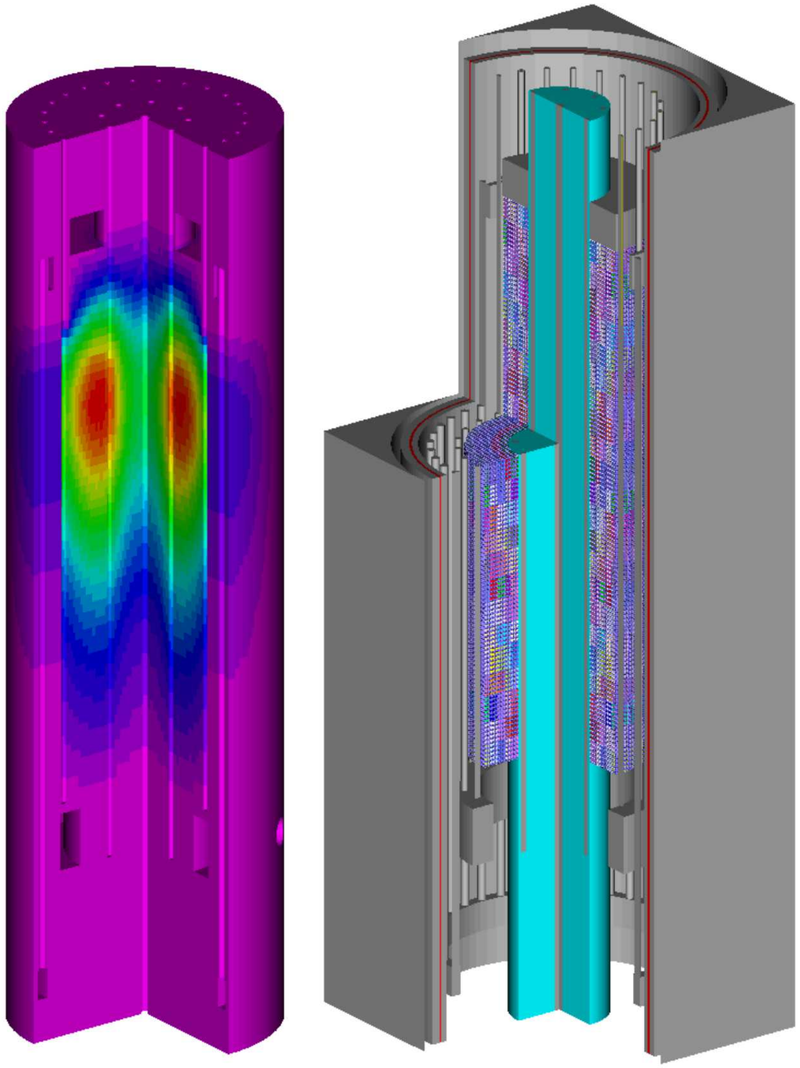

U.S.NRC
United States Nuclear Regulatory Commission
Protecting People and the Environment

NUREG/CR-7107
ORNL/TM-2011/161

Validation of SCALE for High Temperature Gas-Cooled Reactor Analysis

Office of Nuclear Regulatory Research

- Design features
  - Fueled by graphite pebbles containing UO_2-bearing TRISO fuel particles
  - Pebbles circulate multiple passes through the core to high discharge burnup
- Modeling strategy
  - Analysis focused on understanding axial & radial power shape, neutron spectrum for depletion calculations
  - Facilitate depletion calculations via pre-calculated Origen reactor data libraries
- Radionuclide inventory and decay heat data provided for MELCOR model



PBMR-400 SCALE geometry & neutron flux profile

PBMR-400 – Demo Calculations

Depressurized loss-of-forced circulation (DLOFC) accident

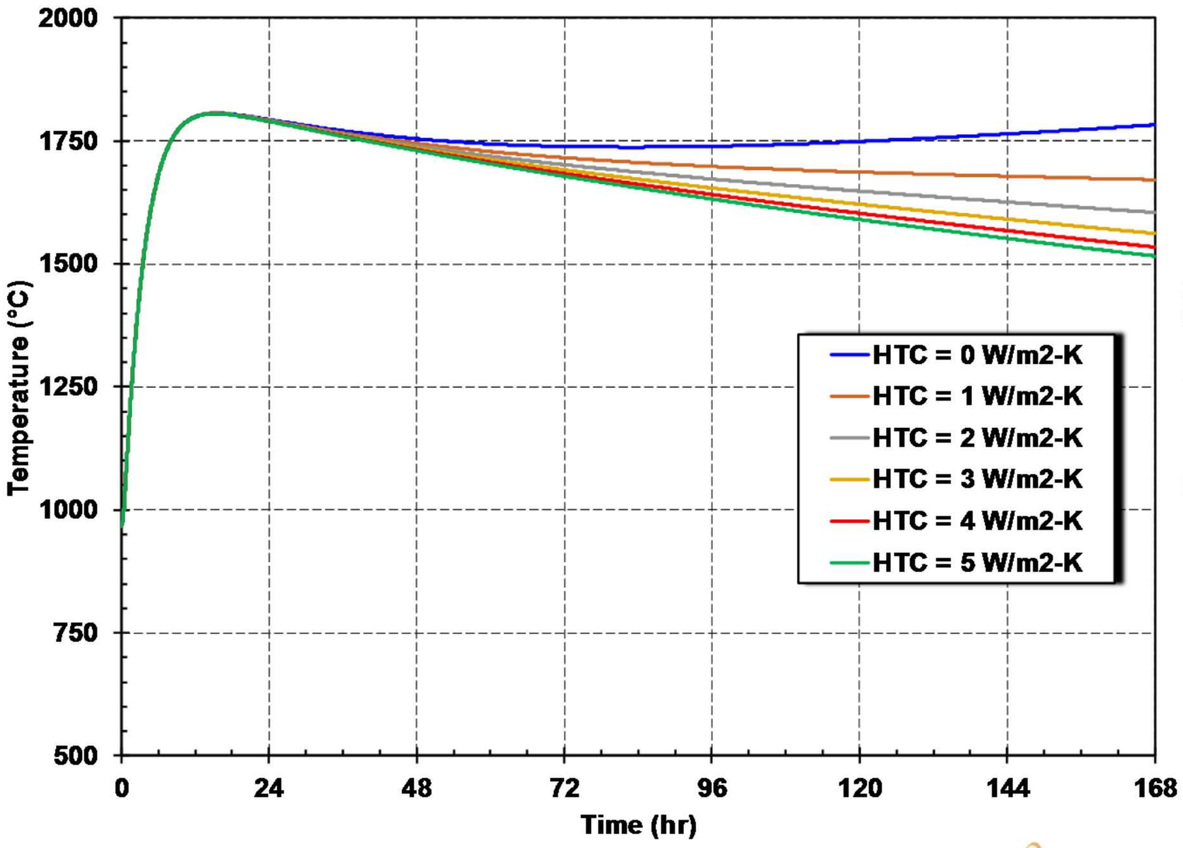
- Large recirculation pipe break
  - Reactor trip
  - Secondary system trips & isolates
  - Passive reactor cavity cooling system (RCCS) available
- Reference case includes nominal heat transfer from the RCCS
- Vessel to RCCS heat transfer sensitivity
  - Heat transfer coefficient to air in the RCCS varied from 0 to 5 W/m^2 K
- RCCS blockage sensitivity
  - Natural circulation air flow area into the RCCS decreased by 90% and 99%

TRISO fission product release model

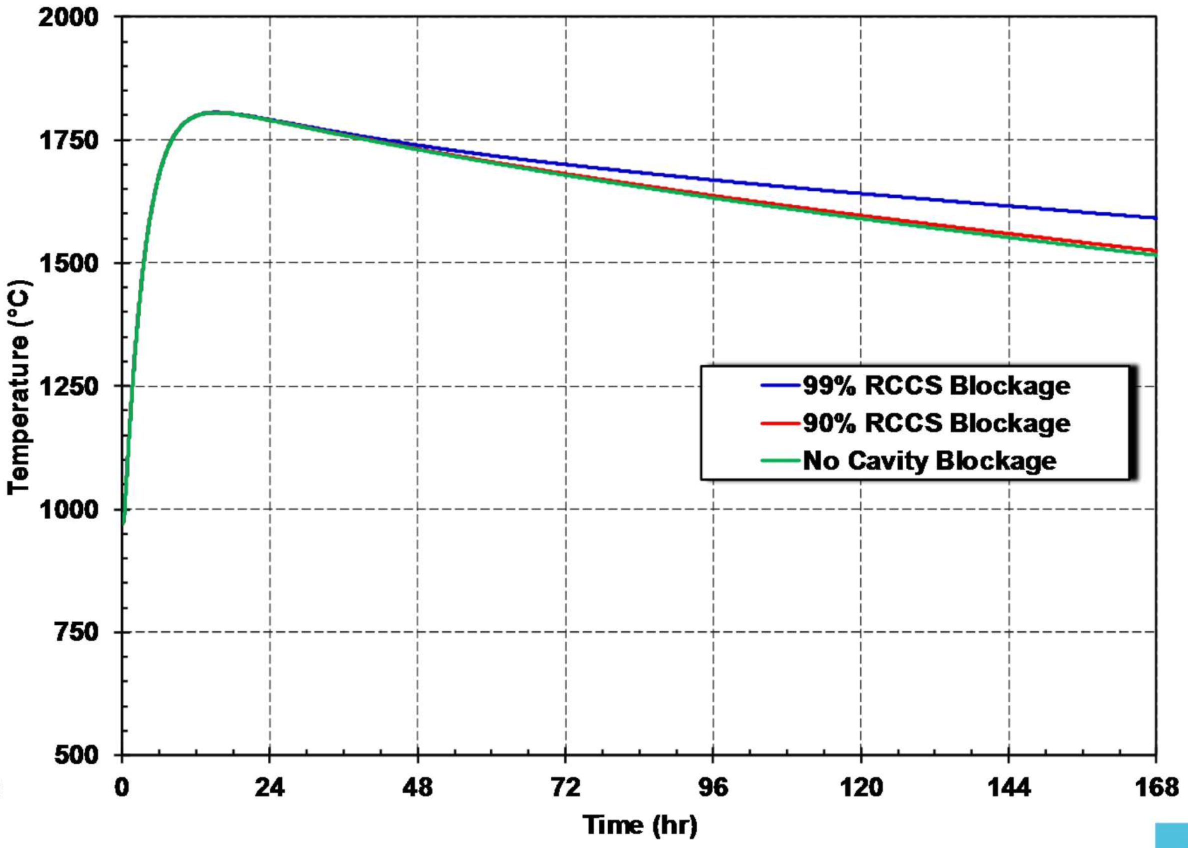
- Diffusivity data from IAEA TECDOC-978, Appendix A
- Fuel failure fraction is user-specified – temperature dependent curve

PBMR-400 – DLOFC results

Peak fuel temperature sensitivity to the RCCS heat transfer coefficient

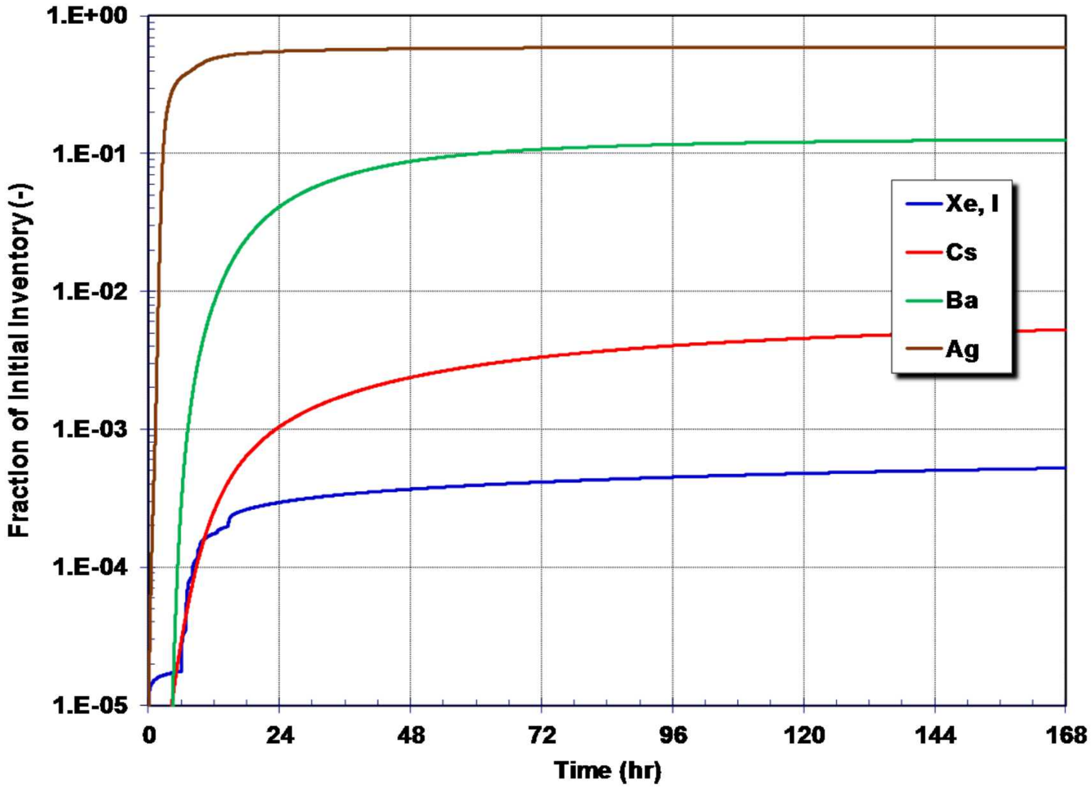


Peak fuel temperature sensitivity to RCCS blockage

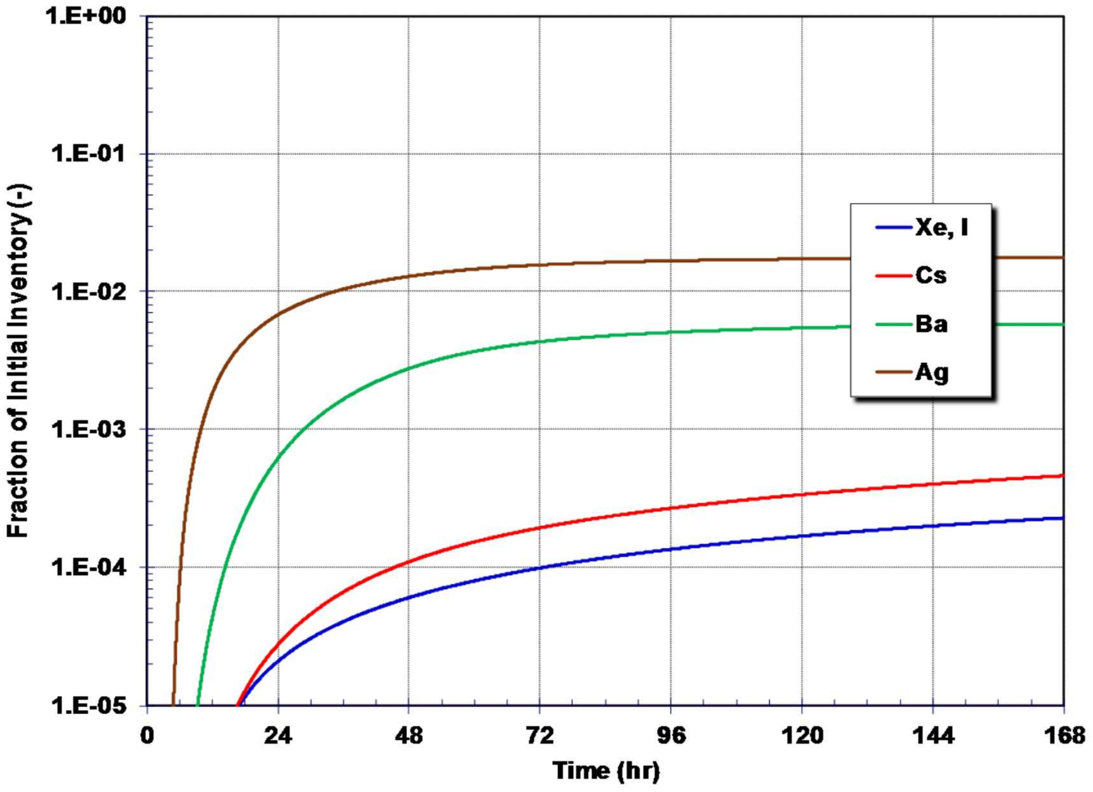


PBMR-400 – DLOFC reference case results

Release from the pebbles to the coolant



Release to the environment



Fluoride-Salt-Cooled High-Temperature Reactor (FHR)

FHR Fluoride-Salt-Cooled, High-Temperature Reactor

Mark-1 PB-FHR Technical Description

Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant

Charalampos "Harry" Andreades
Anselmo T. Cisneros
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Massimiliano Fratoni
Sea Hong
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David L. Krumwiede
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UCBTH-14-002

September 30, 2014
Department of Nuclear Engineering
University of California, Berkeley

This research is being performed using funding received from the U.S. Department of Energy Office of Nuclear Energy's Nuclear Energy University Programs.

NEUP Nuclear Energy University Programs
U.S. Department of Energy

Technical Description of the Mark-1 PB-FHR Power Plant 1 | 153

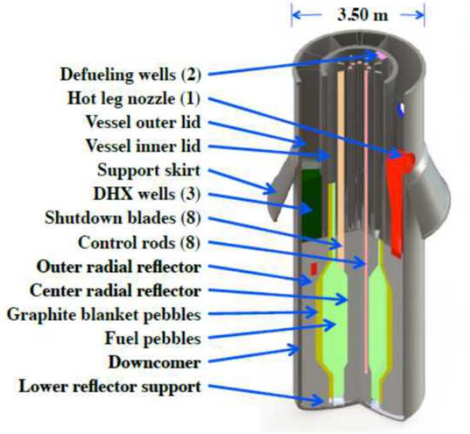


Figure 1-3. The Mk1 PB-FHR reactor vessel

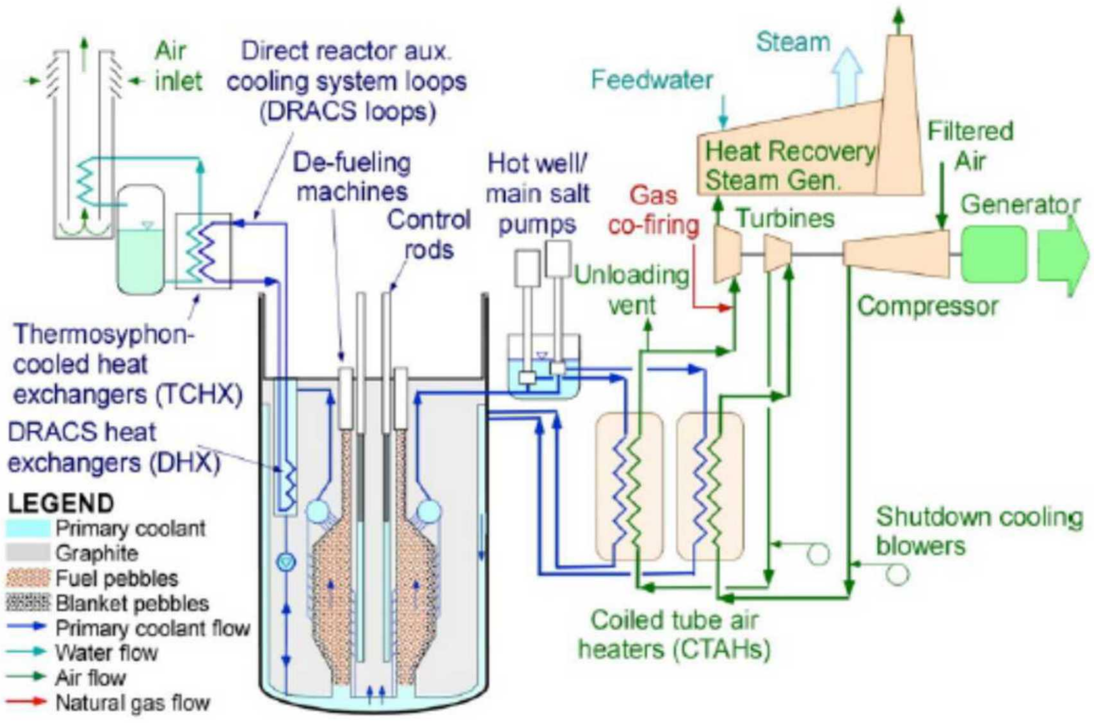
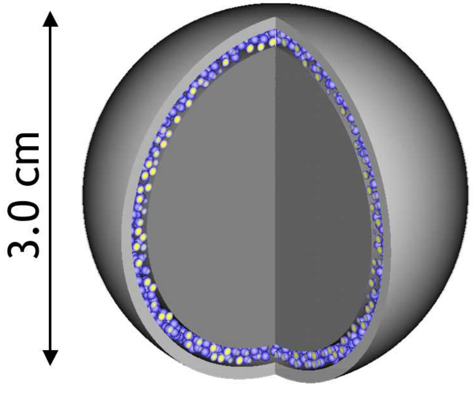


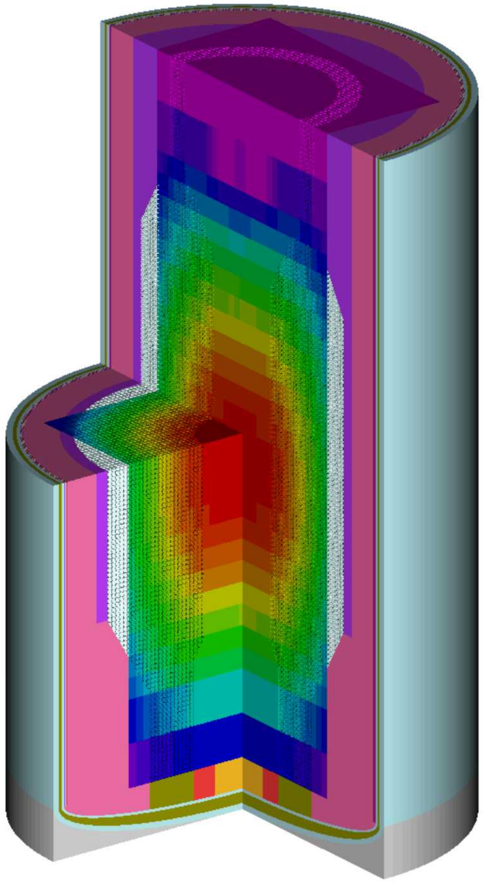
Figure 1-1. Mk1 PB-FHR flow schematic.

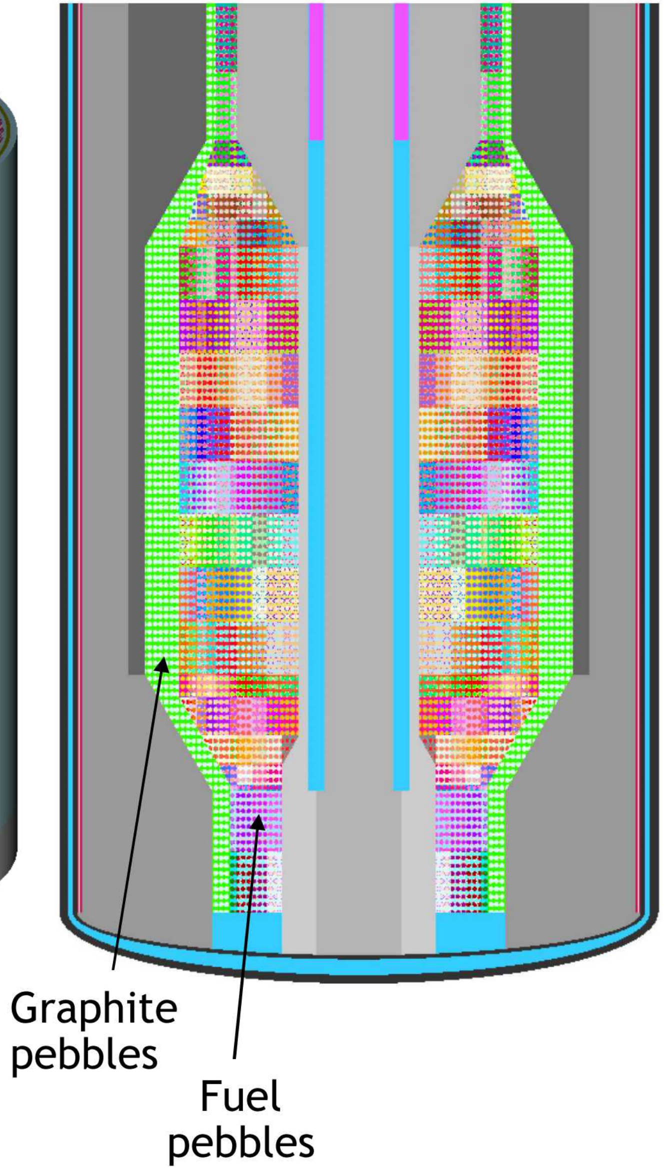
FHR SCALE model

- Design features
  - TRISO particles with UCO fuel kernel (19.9% ^{235}U) in graphite pebbles
  - 236 MW_{th} core with approx. 470,000 fuel pebbles & 218,000 graphite pebbles
  - FLiBe salt coolant
- Modeling strategy
  - Fixed pebble positions (no buoyancy effects)
- Radionuclide inventory and decay heat data provided for MELCOR model



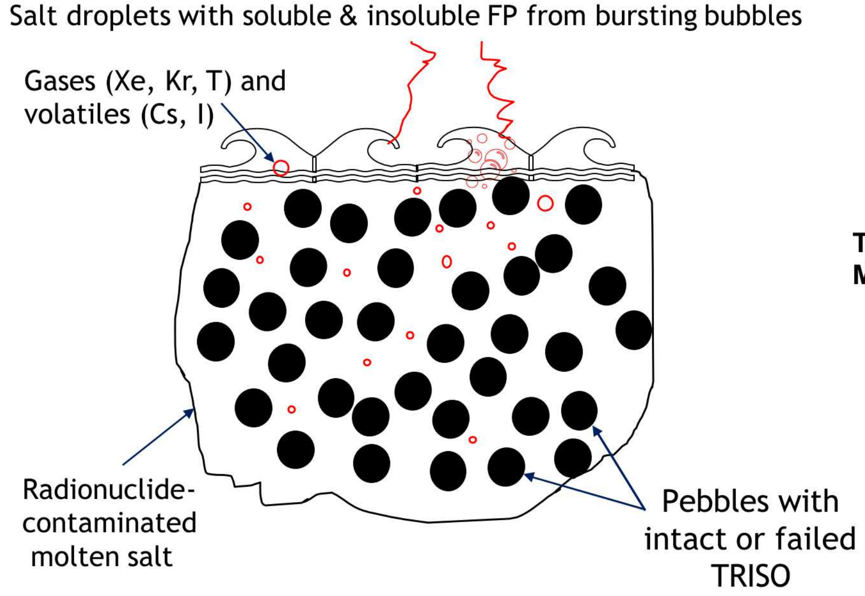
Fuel pebble



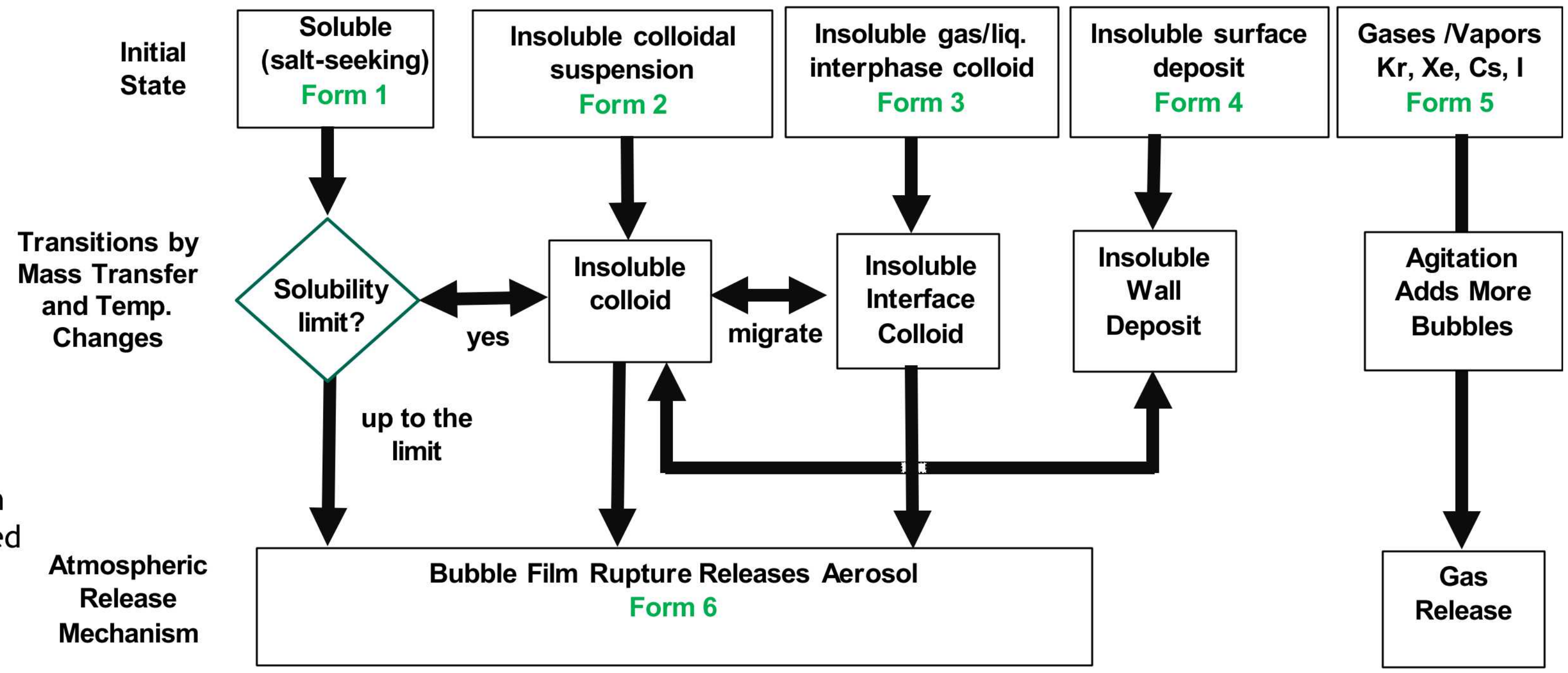


MELCOR fission product release model

Vaporization and bubble burst release (see Vol. 3)



Radionuclides grouped into 6 forms as found in the Molten Salt Reactor Experiments at ORNL



Concluding remarks and next steps

Preliminary working input models

- INL Design A - November 2020
- PBMR-400 - November 2020
- FHR model - March 2021

Followed by public workshops

New computer code versions will be released with updated phenomenological models