SAND2022-12412 PE

SCALE/MELCOR Non-LWR Source Term Demonstration Project – Sodium Fast Reactor (SFR)

September 20, 2022

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Outline

NRC strategy for non-LWR source term analysis

Project scope

Overview of Sodium Fast Reactor (SFR)

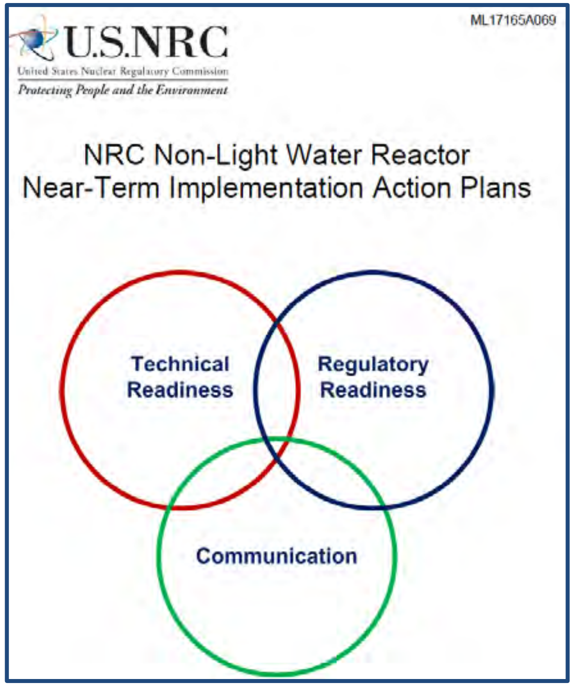
SFR reactor fission product inventory/decay heat methods & results

MELCOR SFR model

SFR plant model and sample analysis

Summary

Integrated Action Plan (IAP) for Advanced Reactors



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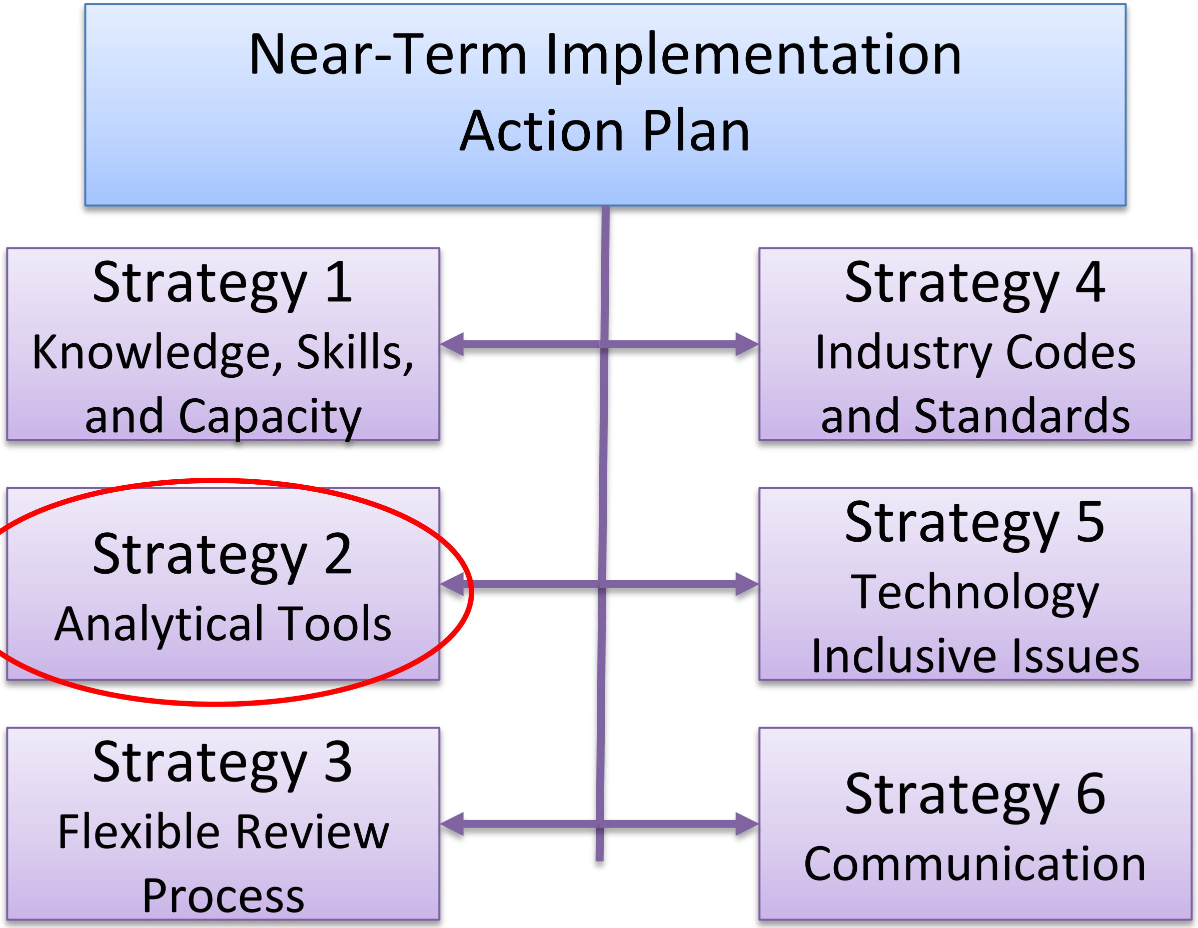
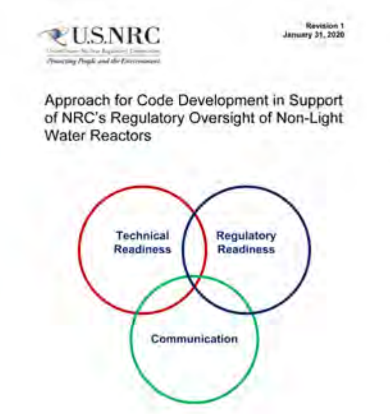


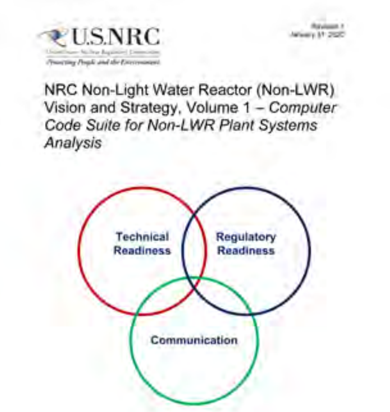
IAP Strategy 2 Volumes


U.S.NRC

Revision 1
January 31, 2020

Approach for Code Development in Support of NRC's Regulatory Oversight of Non-Light Water Reactors

Introduction
ML20030A174


U.S.NRC

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 1 – *Computer Code Suite for Non-LWR Plant Systems Analysis*

Volume 1
ML20030A176


U.S.NRC

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 2 – *Fuel Performance Analysis for Non-LWRs*

Volume 2
ML20030A177


U.S.NRC

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 4 – *Licensing and Siting Dose Assessment Codes*

Volume 4
ML21085A484

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.

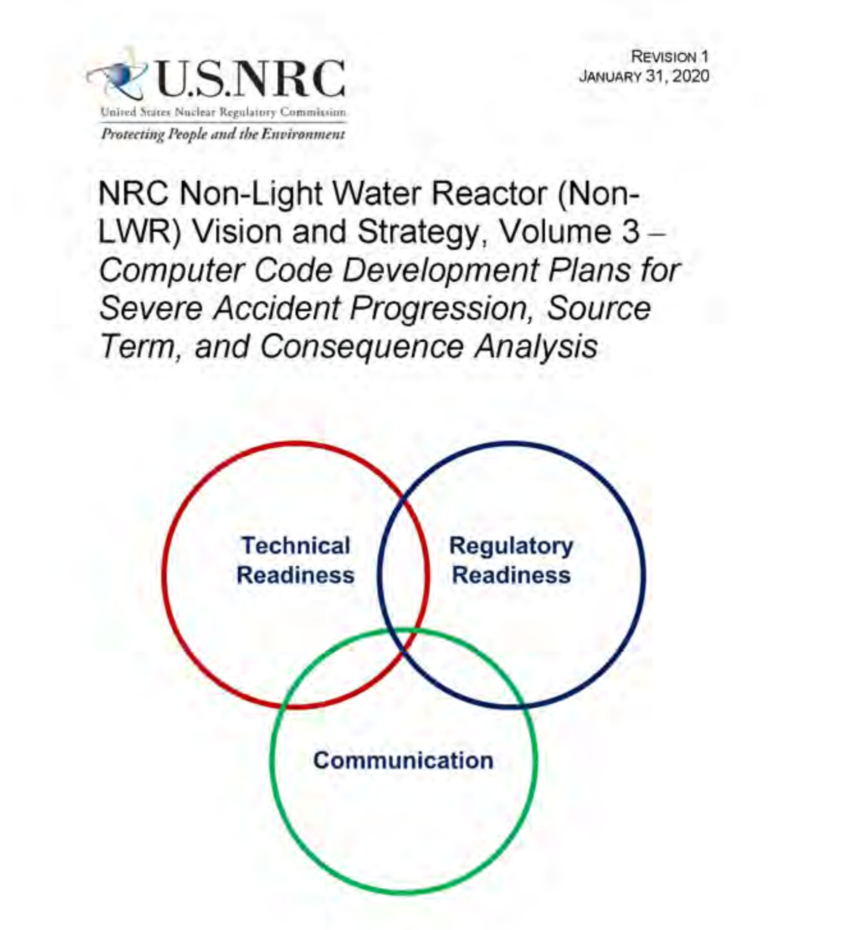

U.S.NRC

Protecting People and the Environment

Revision 1
March 31, 2021

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Volume 5
ML21088A047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United States Nuclear Regulatory Commission

Protecting People and the Environment

Revision 1
January 31, 2020

NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

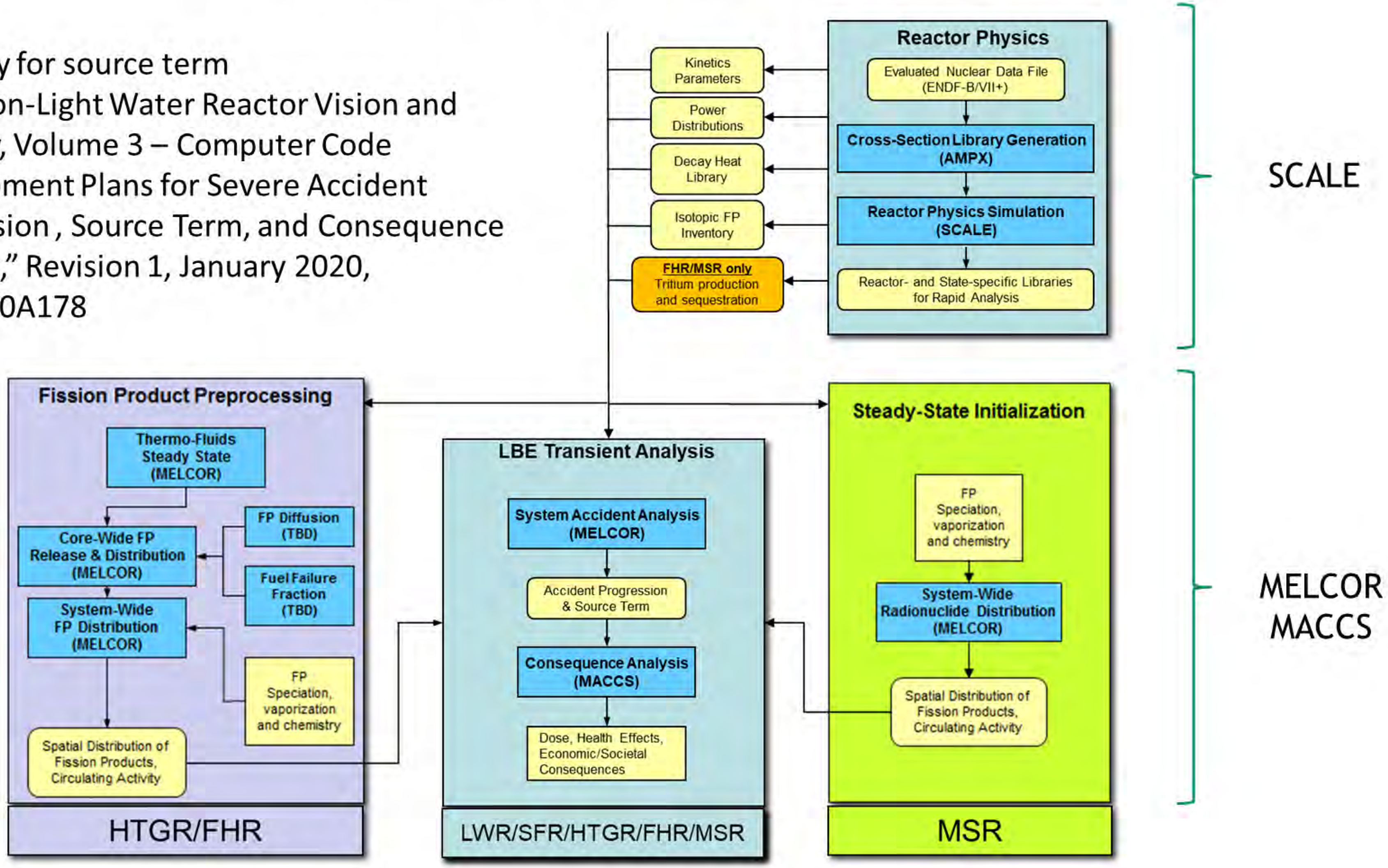
Volume 3
ML20030A178

NRC strategy for non-LWR analysis (Volume 3)

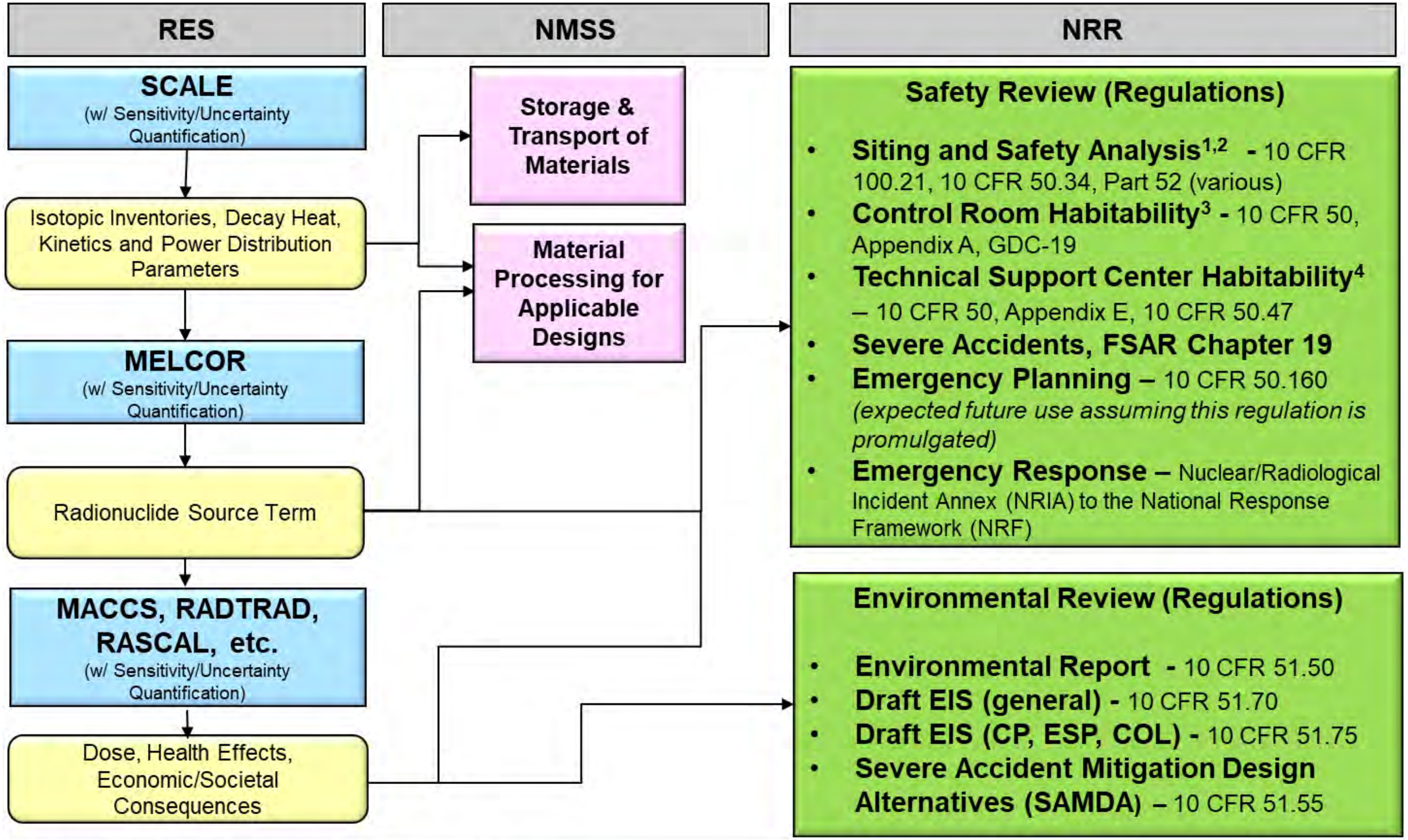
Evaluation Model and Suite of Codes

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 and sample calculations for representative non-LWRs

2021

- Heat pipe reactor – INL Design A
- Pebble-bed gas-cooled reactor – PBMR-400
- Pebble-bed molten-salt-cooled – UCB Mark 1
- Public workshop videos, slides, reports at advanced reactor source term webpage

2022

- Molten-salt-fueled reactor – MSRE – public workshop 9/13/2022
- Sodium-cooled fast reactor – ABTR – public workshop 9/20/2022

2023

- Additional code enhancements, sample calculations, and sensitivity studies

Project approach

1. Build SCALE core model and MELCOR full-plant model
2. Select scenarios that demonstrate code capabilities
3. Perform simulations
    - Use SCALE to model decay heat, core radionuclide inventory, and reactivity feedback
    - Use MELCOR to model accident progression and source term
    - Perform sensitivity cases

Sodium Fast Reactor (SFR)
(US History)

Sodium fast reactors (1/4)

Experimental Breeder Reactor 1

- Object to prove Enrico Fermi's fuel breeding principle and to generate electricity
  - Construction started in 1949
  - Uranium metal plate fueled with liquid sodium-potassium (NaK) coolant
  - 1.4 MW_{th} (200 kW_e)and generated enough electricity to run four 200-W light bulbs (Dec 1951)

EBR Unit 1

[https://en.wikipedia.org/wiki/Experimental_Breeder_Reactor_I#/media/File:First_four_nuclear_lit_bulbs.jpeg]

Experimental Breeder Reactor 2

- Demonstrate a complete breeder reactor nuclear power plant
  - Construction started in 1964 and reached full power in 1969 - *62.5 MW_{th} and 20 MW_e*
- Used uranium metal fuel rods
  - Fuel designs researched and refined
  - 96,399 uranium metal fuel slugs fabricated with 35,000 irradiated
- Demonstrated passive safety tests
  - Unprotected loss of flow
  - Unprotected loss of heat removal

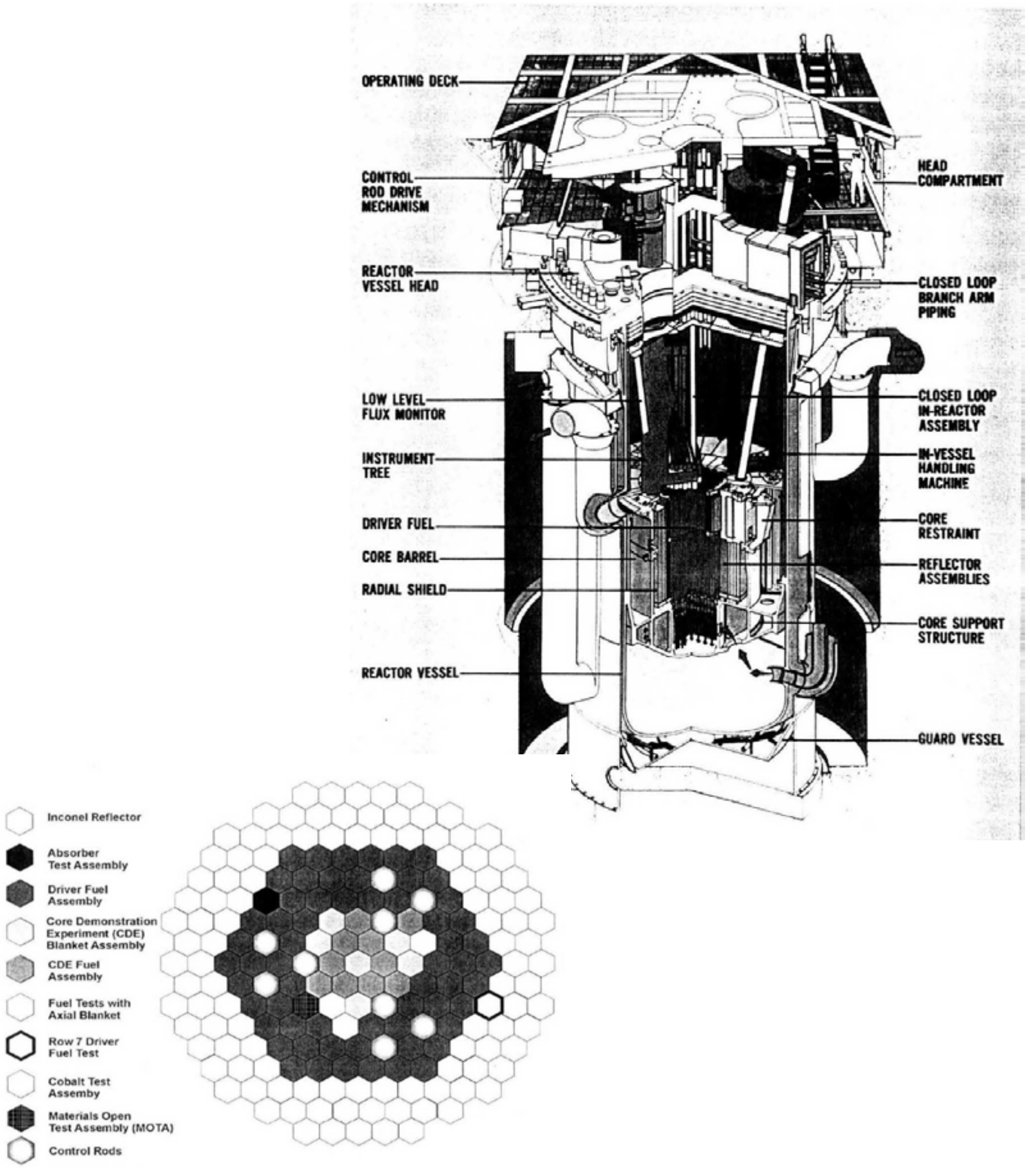
EBR Unit II

[https://www.ne.anl.gov/About/reactors/ebr2/EBRII_hirez.jpg]

Sodium fast reactors (2/4)

Fast Flux Test Facility (FFTF) reactor

- Design started late 1970s and first criticality in 1982
  - 400 MW_{th} power rating
  - National testing and research facility for advanced nuclear fuels, material, component, and passive safety features
  - Generated medical isotopes and tritium for the US fusion program
  - Overlapped EBR-2 and operated for ~10 years
- Used mixed oxide metal fuel design
  - Same construction as EBR-2
  - 40,000 fuel pins irradiated with only one fuel pin cladding failure
  - Goal burn-up to 100 GWd/MTM (achieved maximum burn-up of 238 GWd/MTM)
- Important testbed for instrumentation and safety features
  - Instrumentation to verify natural circulation
  - Guard vessel around the reactor vessel to contain sodium spills

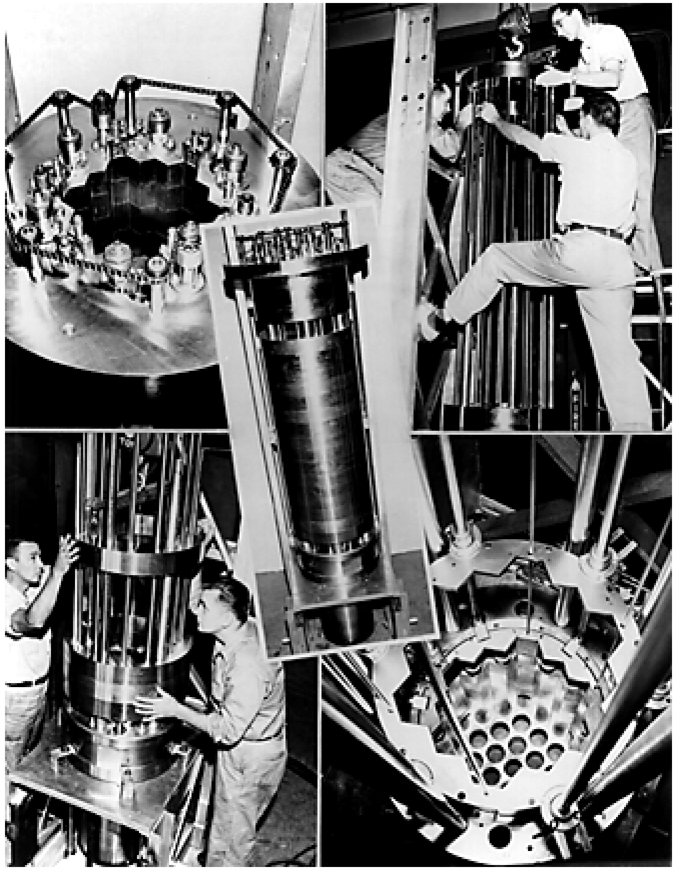


FFTF Reactor Core and Vessel
[FFTF-20083, Rev 0]

Sodium fast reactors (3/4)

Fermi 1 - Prototype breeder reactor (200 MW_{th} and 68 MW_e)

- Construction began in 1956 & operated from 1963 to 1972
- Used uranium metal fuel (26% enriched U-235 fuel)
  - 92 fuel assemblies surrounded by 548 fuel assemblies of depleted uranium
- Exhibited coolant flow blockage on October 5, 1966
  - Zr plate, near the bottom of the reactor, became loose and blocked the inlet nozzles – restricted sodium coolant flow
  - 2 damaged fuel assemblies – resulting in partial fuel melts
  - No radionuclide release to the environment but Fermi 1 underwent an extended shutdown for clean-up and repairs
  - Restarted and ran from 1970 to 1972



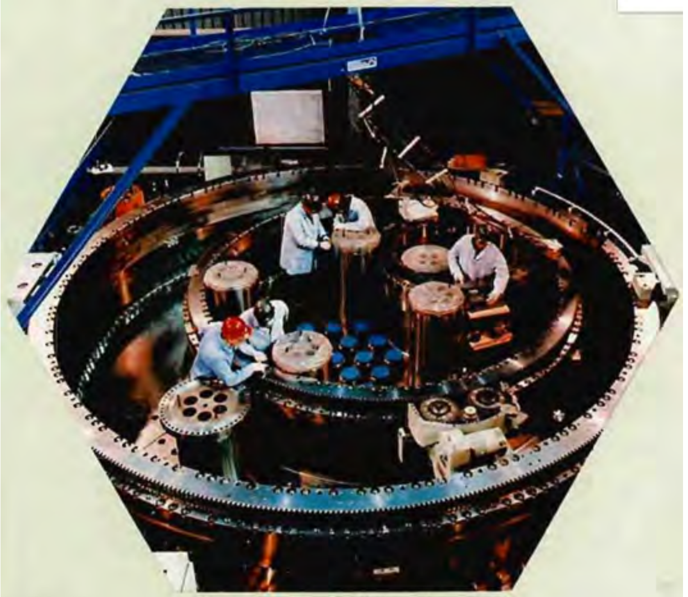
Fermi Unit 1
[https://en.wikipedia.org/wiki/Enrico_Fermi_Nuclear_Generating_Station]

Sodium fast reactors (4/4)

Clinch River Liquid Metal Fast Breeder Reactor Project

- Authorized in 1970
  - 1000 MW_{th} and 350 MW_e
  - Mixed oxide (plutonium & uranium) fuel in 108 fuel assemblies
- Stimulated advances in research, design, component fabrication, safety analysis, and licensing
  - Fabrication of $380M of major components delivered (~50% of planned components)
- Licensing activities started in 1974
  - Environmental Impact Statement approved in 1977
  - ASLB issued memorandum of findings in 1984 that all issues related to the construction permit had been addressed
  - 250,000 pages of documentation for the licensing effort
- Project terminated in 1983
  - DOE concluded the project demonstrated the ability to license LMBRs

Prototype sodium pump internals

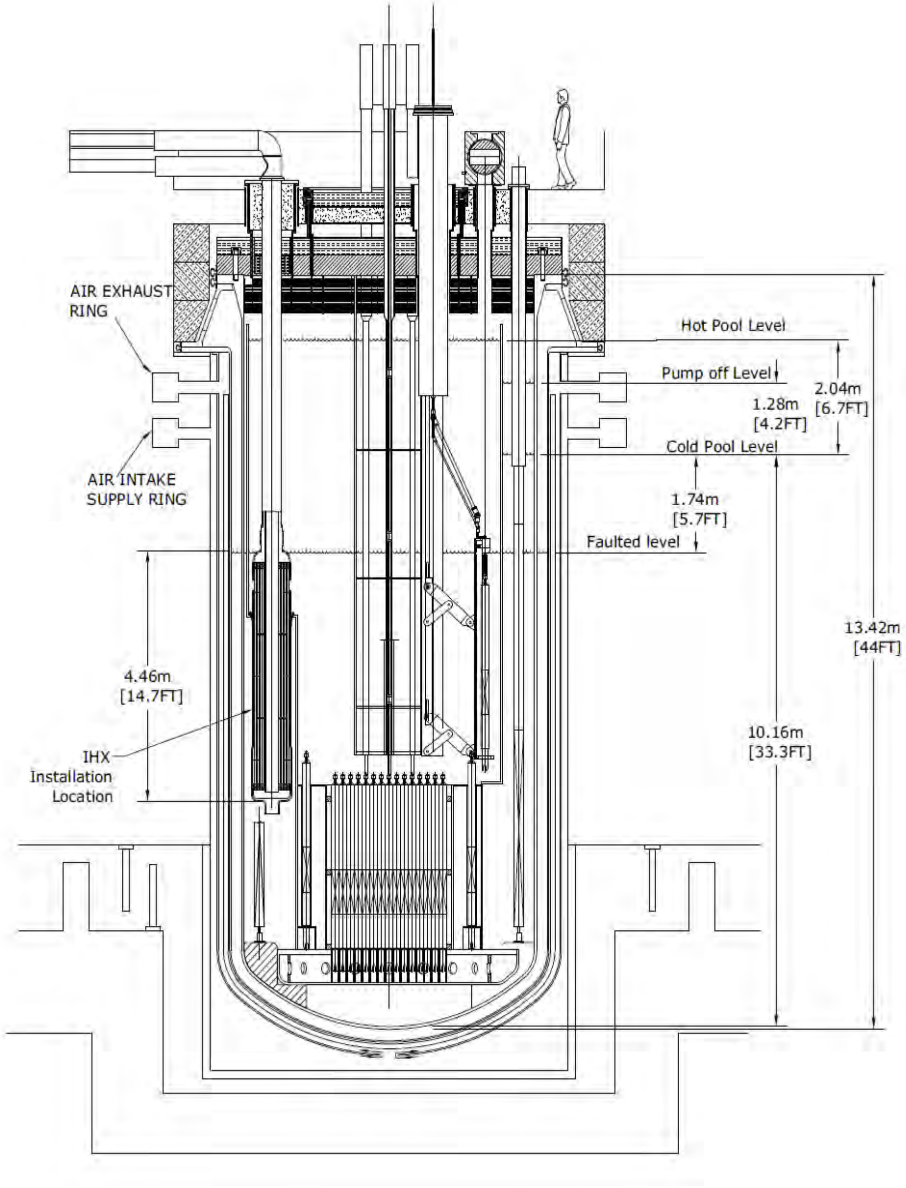


Reactor vessel closure head

Clinch River Project [https://www.nrc.gov/docs/ML1806/ML18064A893.pdf]

ABTR – Reactor Design

- Selected for the SCALE/MELCOR SFR demonstration
- ABTR Design Specifics
  - 250 MW_{th}
  - Pool-type SFR, near atmospheric pressures
  - 355°C core inlet / 510°C core outlet
  - 1260 kg/s core flowrate
  - 2 mechanical or EM pumps
  - 2 internal intermediate heat exchangers
- Design features
  - Guard vessel
  - Short-term fuel storage in the reactor
  - Primary connects to an intermediate loop inside the vessel



ABTR Vessel
[ANL-AFCI-173]

ABTR core

- 199 hex assemblies
  - HT-9 steel duct surrounds each assembly
  - Small interstitial gap region between assemblies
- Multiple assembly type and region core
  - 24 inner core driver assemblies
  - 30 outer core driver assemblies
  - 6 fuel test locations
  - 10 control assemblies (B_4C)
  - 3 material test assemblies
  - 78 reflector assemblies (HT-9 pins)
  - 48 shield assemblies (B_4C)
- Color coding identifies diverse functions and assembly materials

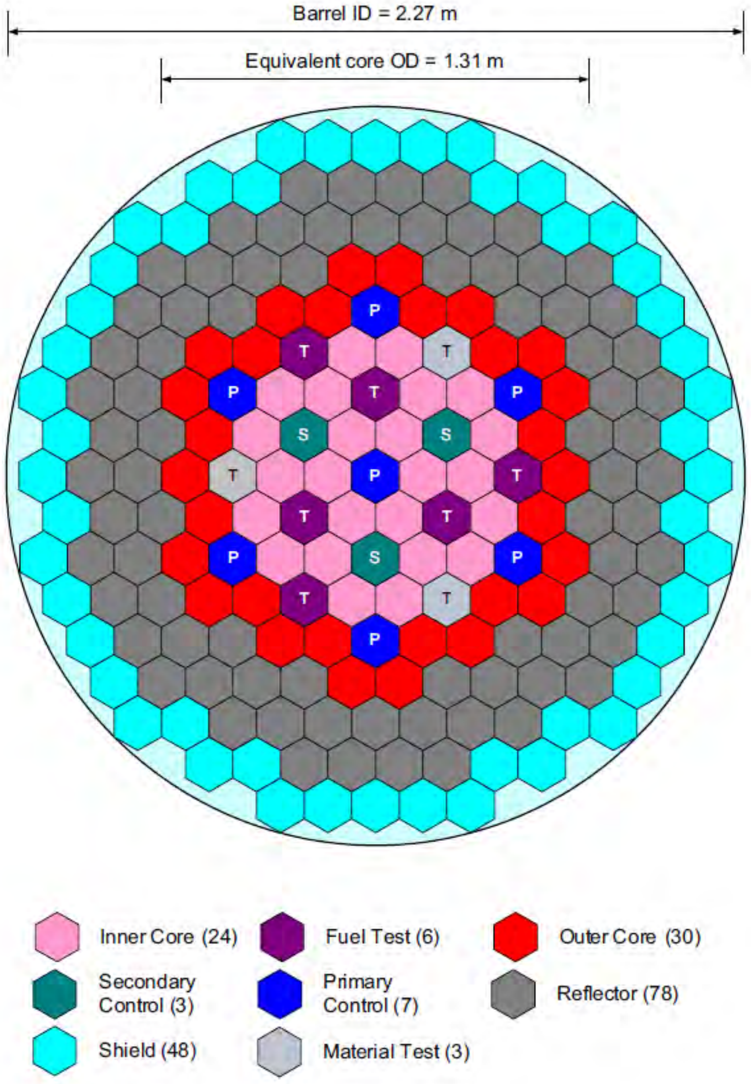
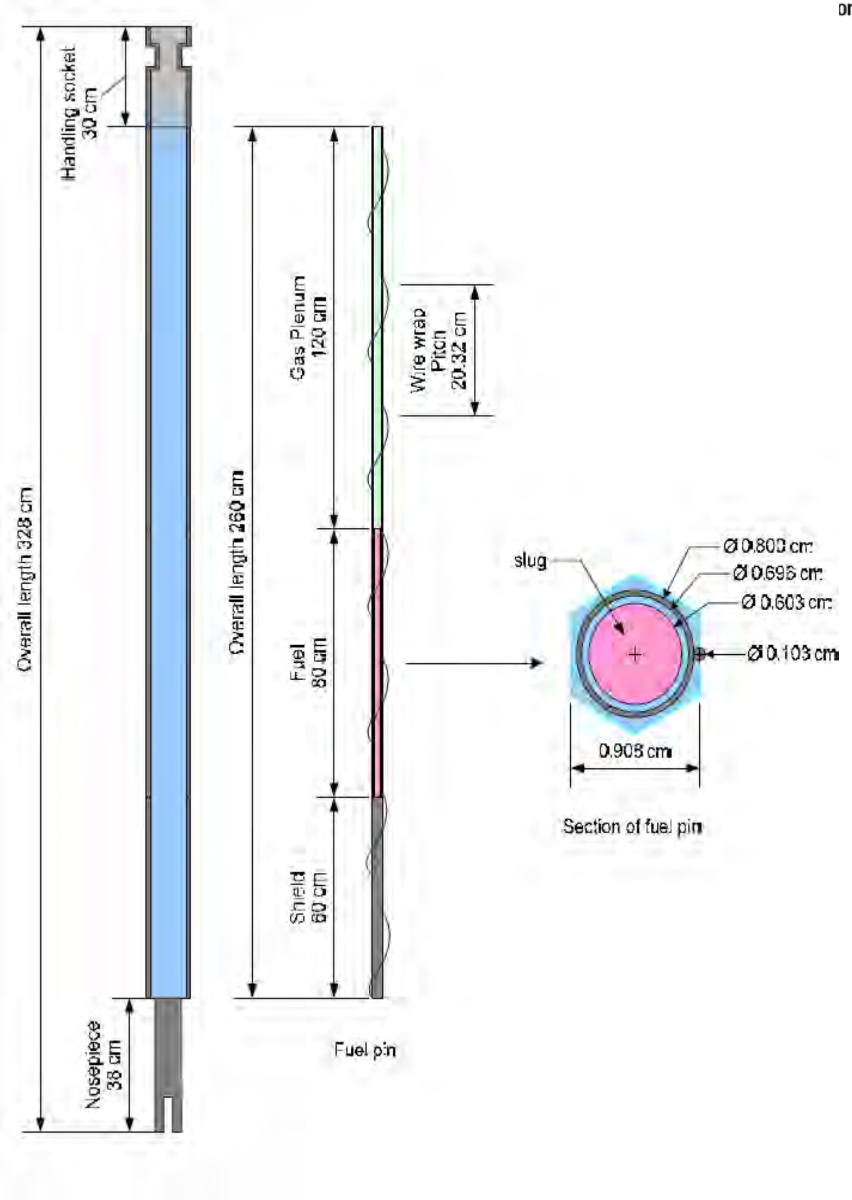


Figure II.1-1 Reference ABTR Core Configuration

ABTR Vessel
[ANL-AFCI-173]

ABTR fuel

- A hex HT-9 alloy duct surrounds 217 fuel rods
  - HT-9 cladding (melts at 1687 K)
  - Steel wire used to maintain spacing
  - U-TRU-Zr10% metallic fuel
  - 1.2 m argon gas plenum to accommodate expansion and fission gases



ABTR Fuel
[ANL-AFCI-173]

SCALE SFR Inventory, Decay Heat, Power, and Reactivity Methods and Results

U.S.NRC

OAK RIDGE National Laboratory

Sandia National Laboratories

NRC SCALE/MELCOR Non-LWR Demonstration Project

Objectives:

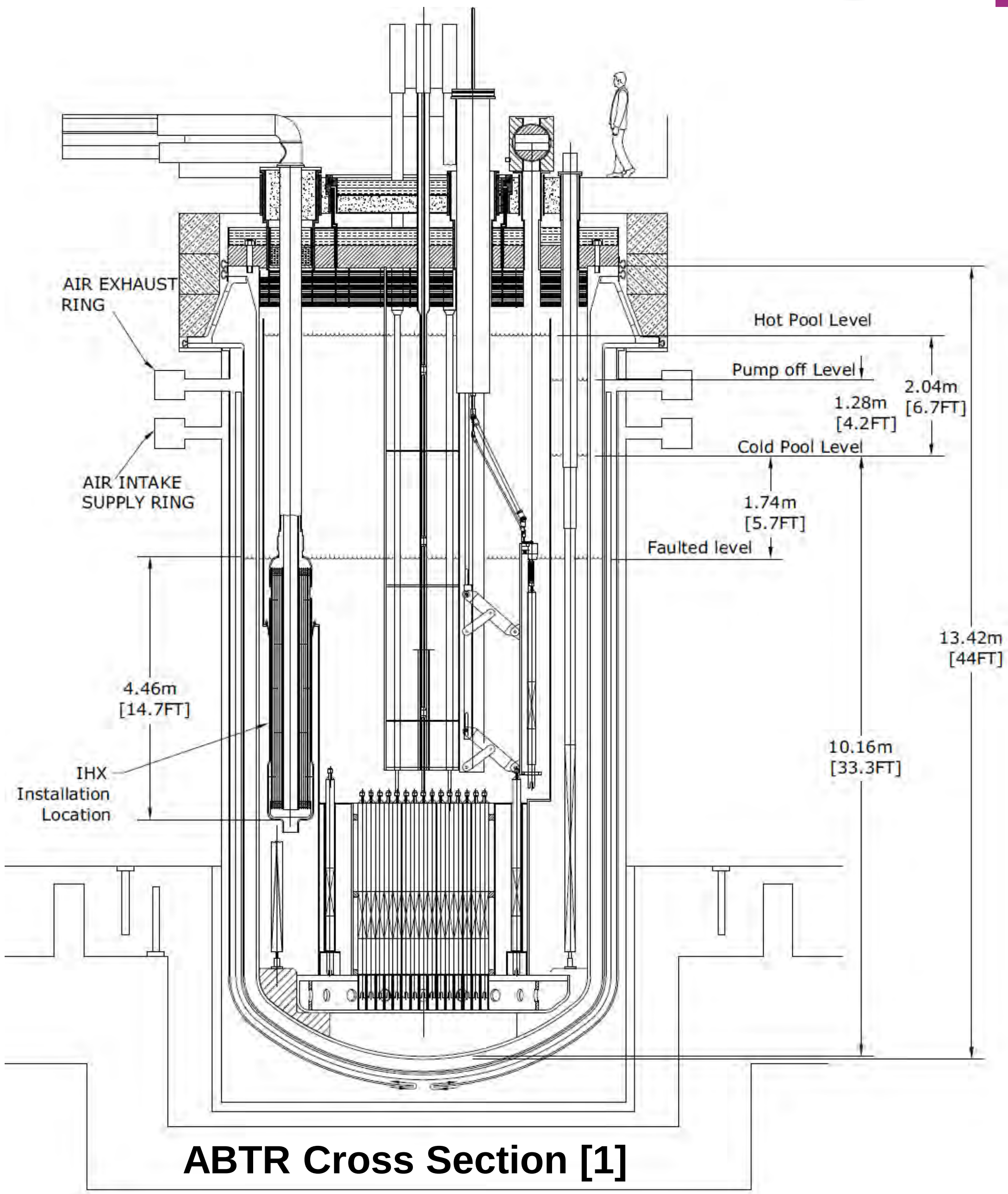
- Develop approach and models for SCALE analysis to obtain:
  - Radionuclide inventory
  - System decay heat
  - Power profiles
  - Reactivity coefficients

Challenges:

- Full core depletion calculation
- Fast neutron spectrum

Approach:

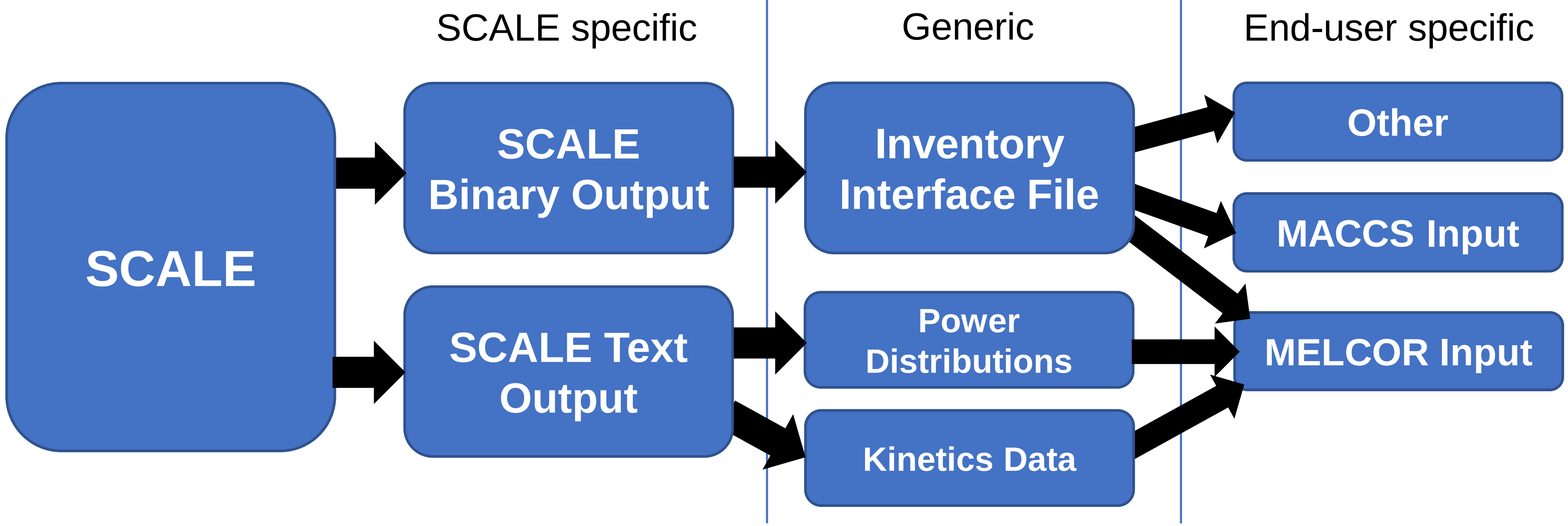
- Develop fully heterogeneous 3D model
- Perform depletion of one cycle
- Evaluate neutronic characteristics
- Verify SCALE results with results in the open literature



ABTR Cross Section [1]

[1] Y. I. Chang, et al. (2006). Advanced Burner Test Reactor Preconceptual Design Report, Technical Report ANL-AFCI 173, Argonne National Laboratory, Argonne, IL.

Workflow

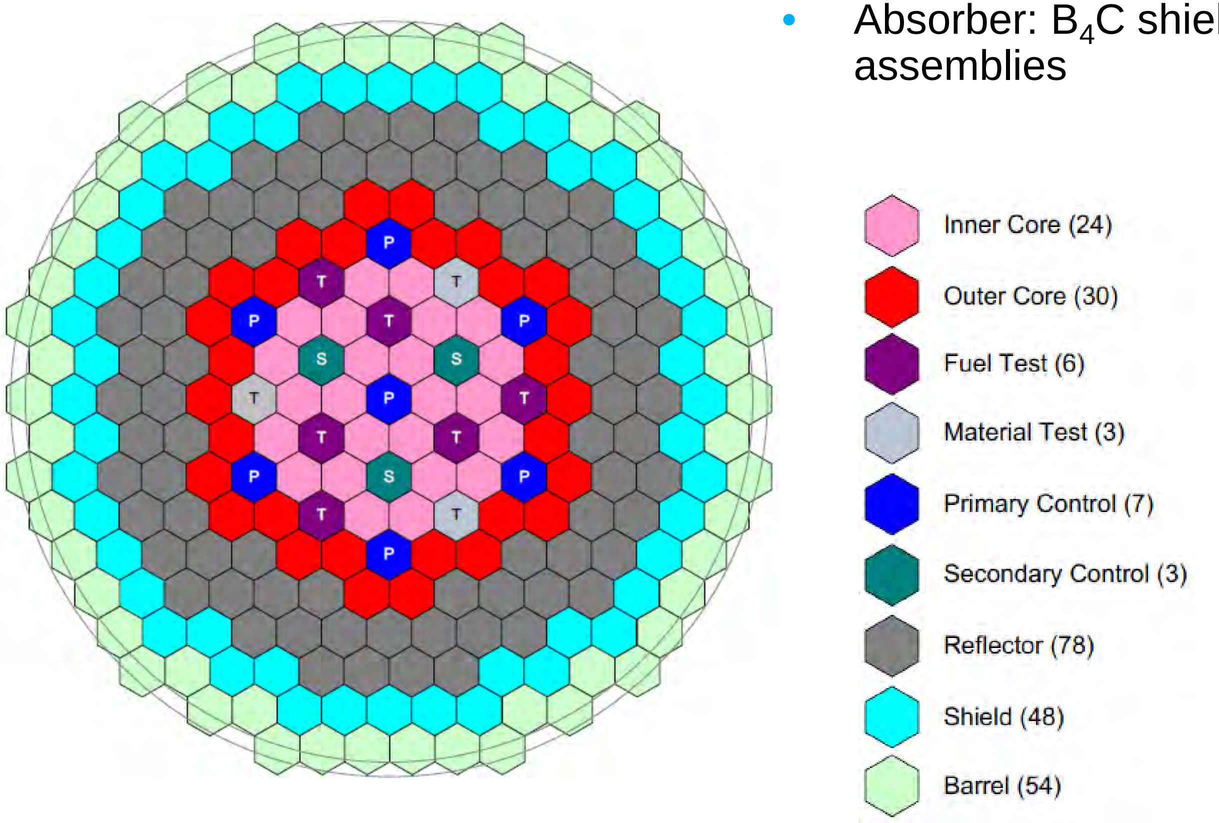


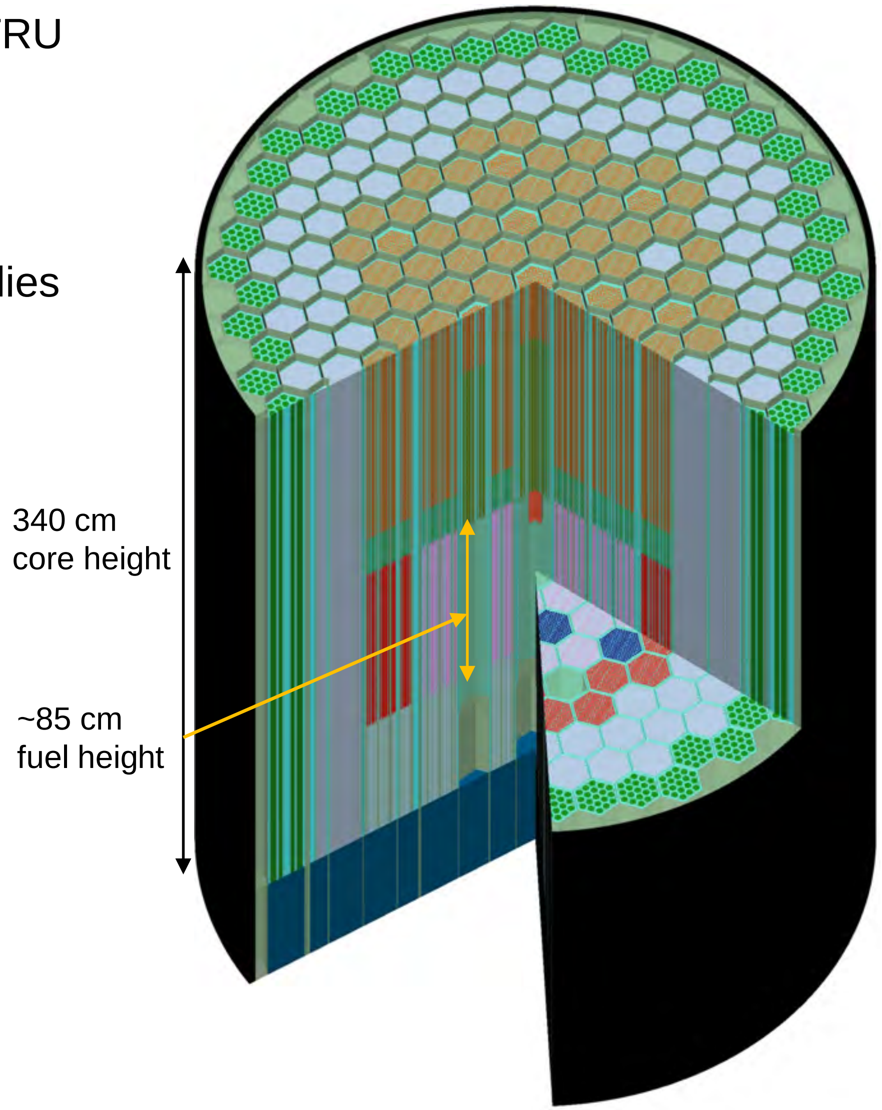
- SCALE capabilities used
  - Codes:
    - KENO-VI 3D Monte Carlo transport
    - ORIGEN for depletion
  - Data:
    - ENDF/B-VII.1 nuclear data library
  - Sequences:
    - CSAS for reactivity (e.g., control assembly worth)
    - TRITON for reactor physics & depletion

ABTR Neutronics Summary

- 250 MWth rated power
- 4-months operating cycle
- Fast spectrum for burning actinides
- 4.05 tHM initial core loading

- Fuel: U/TRU-10Zr (16.5-20.7% TRU content)
- Cladding: HT-9 cladding
- Coolant: sodium
- Reflector: HT-9 reflector assemblies
- Absorber: B_4C shield and control assemblies





3D SCALE Model

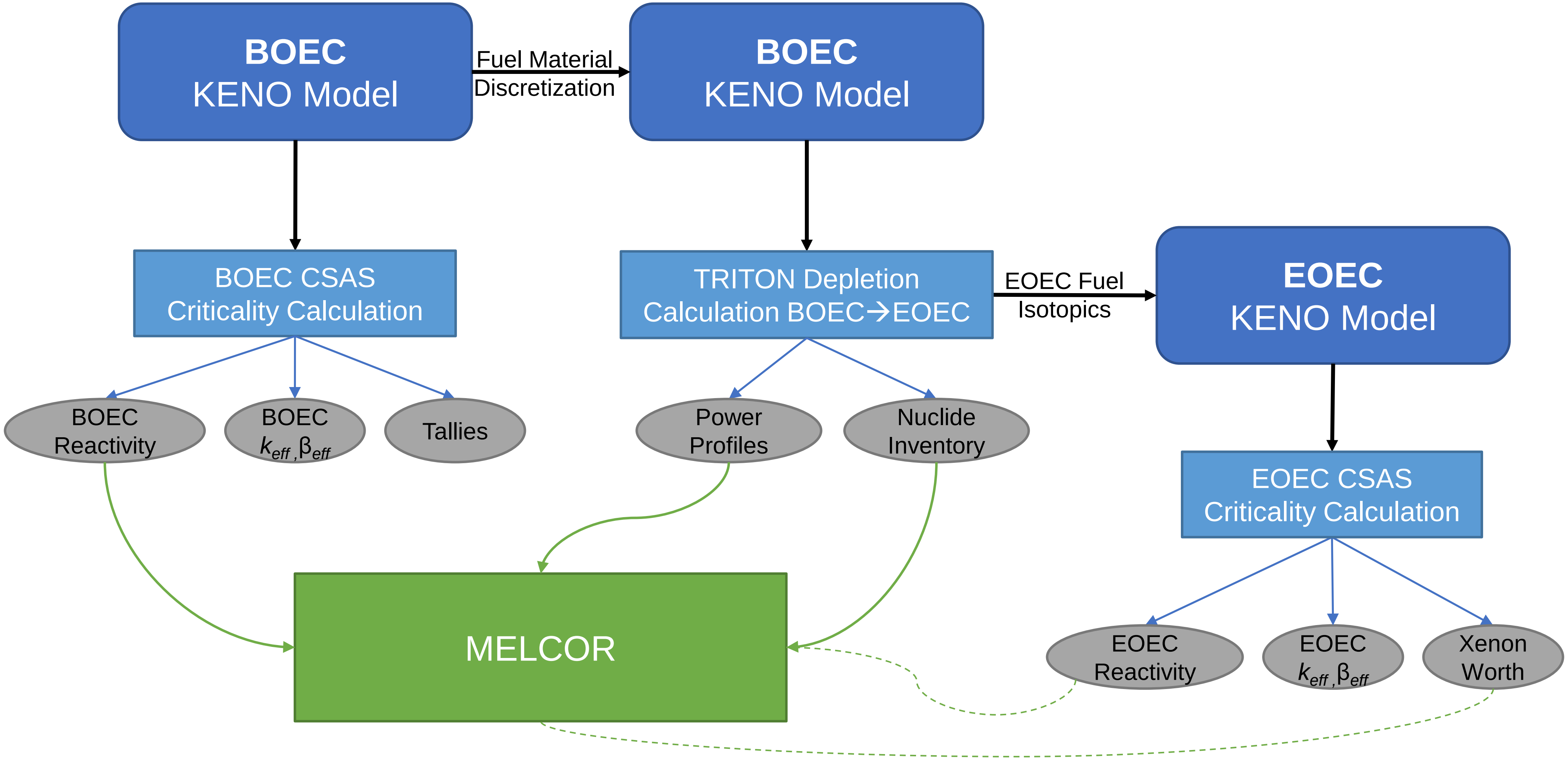
[2] T. K. Kim, "Benchmark Specification of Advanced Burner Test Reactor," ANL-NSE-20/65, Argonne National Laboratory, 2006. doi:10.2172/1761066.

SCALE Analysis Approach

- Develop KENO model of the benchmark:
  - At hot conditions (considering radial and axial expansions as specified by the benchmark)
  - With Beginning of Equilibrium Cycle (BOEC) fuel
  - Simple model for criticality and discretized model for depletion calculation
- Perform CSAS-KENO analysis with simple model at BOEC:
  - Verify eigenvalue (k_{eff}) and effective delayed neutron fraction (β_{eff}) by comparison with ANL ABTR design report [1] and INL publication [3]
  - Analyze 3D flux and fission rate profiles
- Perform TRITON-KENO analysis with discretized model:
  - Deplete model for one cycle to obtain inventory at End of Equilibrium Cycle (EOEC)
  - Analyze reactivity and power profiles at EOEC
- Provide inventories, power profiles, and reactivities to MELCOR
- Perform additional sensitivity studies in support of MELCOR analysis

[1] Y. I. Chang, et al. (2006). Advanced Burner Test Reactor Preconceptual Design Report, Technical Report ANL-AFCI 173, Argonne National Laboratory, Argonne, IL.
[3] C. M. Mueller, et al. (2021). NRC Multiphysics Analysis Capability Deployment FY2021 – Part 2, Technical Report INL/EXT-21-62522, Idaho National Laboratory, Idaho Falls, ID.

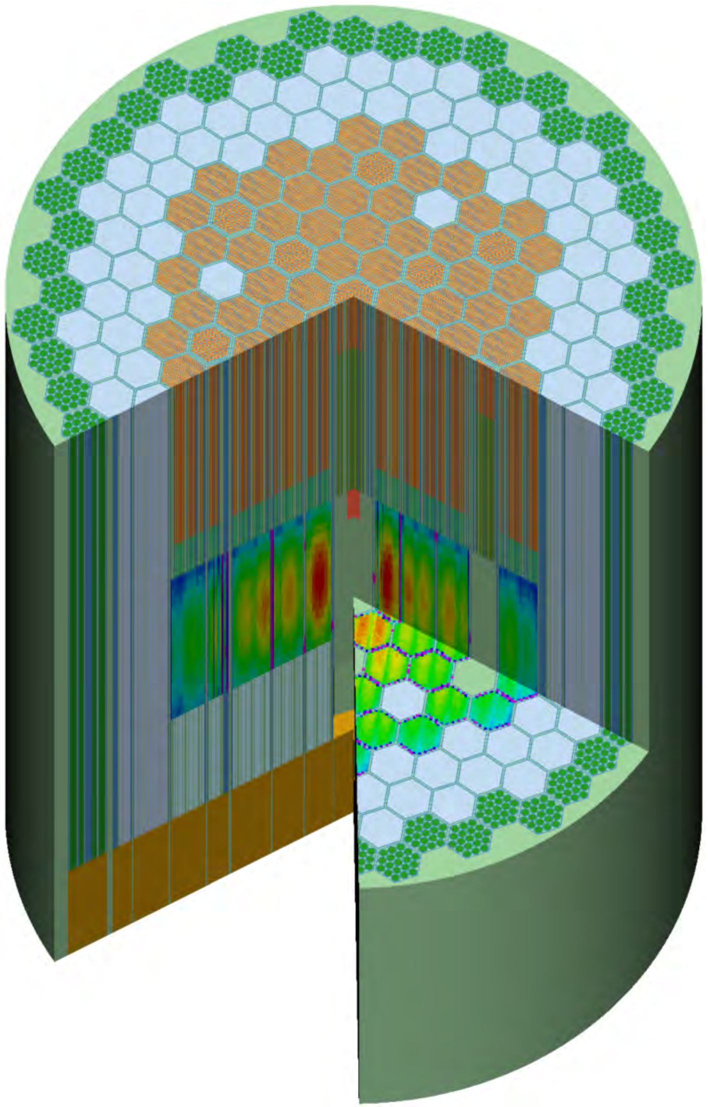
SCALE Analysis Approach



SCALE Model Construction and Verification

Modeling Assumptions

- Full-core 3D Monte Carlo with continuous energy physics
- System state defined in ABTR benchmark specifications [2]
  - BOEC starting isotopics
  - Temperature at hot full power
    - Fuel: 855K
    - Structure: 735K
    - Coolant: 705K
    - Shield: 630K
  - Geometry considers thermal expansion of all components
  - Helium fill gas (assumed)
  - Minor assumptions were made for temperatures not explicitly defined in the benchmark
    - Temperatures are given as a mix of material-specific and region-specific definitions



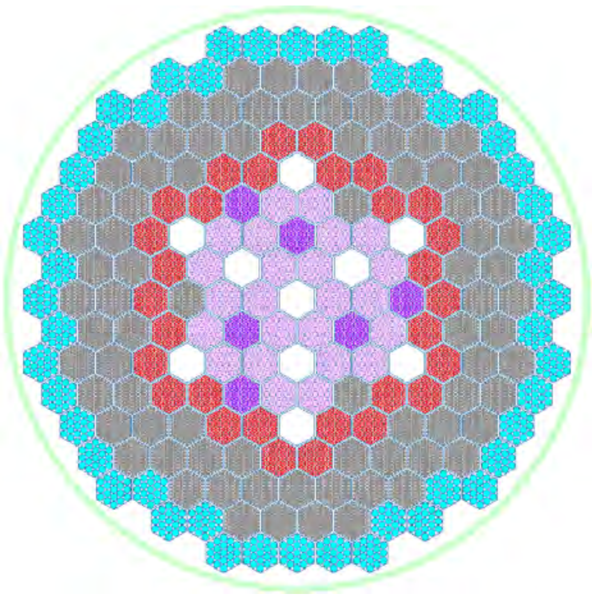
3D SCALE ABTR Core with Fission Density Overlay

[2] T. K. Kim, “Benchmark Specification of Advanced Burner Test Reactor,” ANL-NSE-20/65, Argonne National Laboratory, 2006. doi:10.2172/1761066.

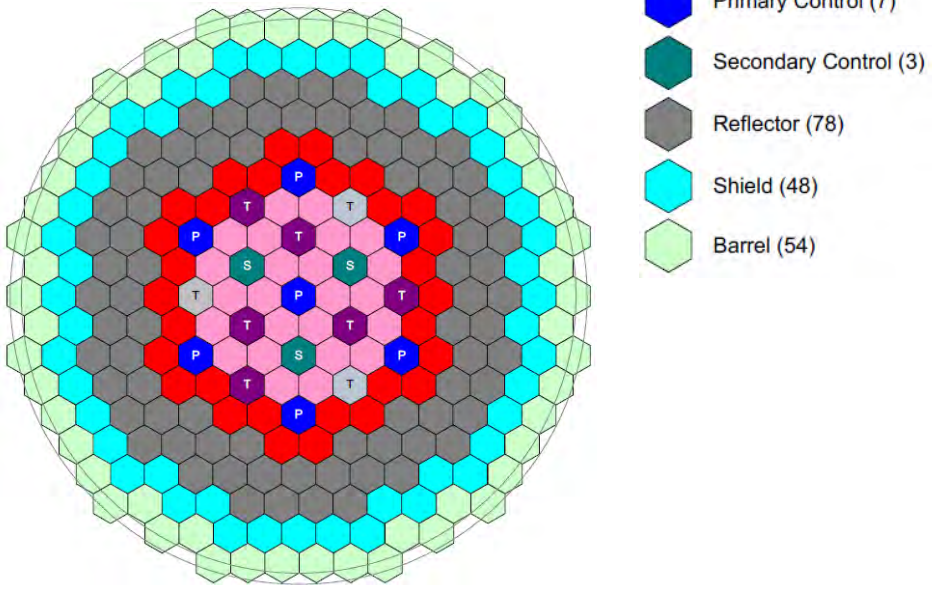
ABTR Model Development

- KENO 3D full core model built based on ABTR benchmark specifications
- Barrel, as described with assemblies, was replaced with a cylindrical configuration
  - Examined 115 and 114.413 cm (expansion of barrel at coolant temperature)
  - Effect is statistically indistinguishable
  - Internal face of the barrel is coolant, while the external face of the barrel is void

Case	k_{eff}	Difference (pcm)
Barrel Assemblies	1.03025 ± 0.00004	(ref)
115 cm Cylinder	1.03017 ± 0.00003	-8.7 ± 5.0
114.413 cm Cylinder	1.03014 ± 0.00004	-11.4 ± 5.0



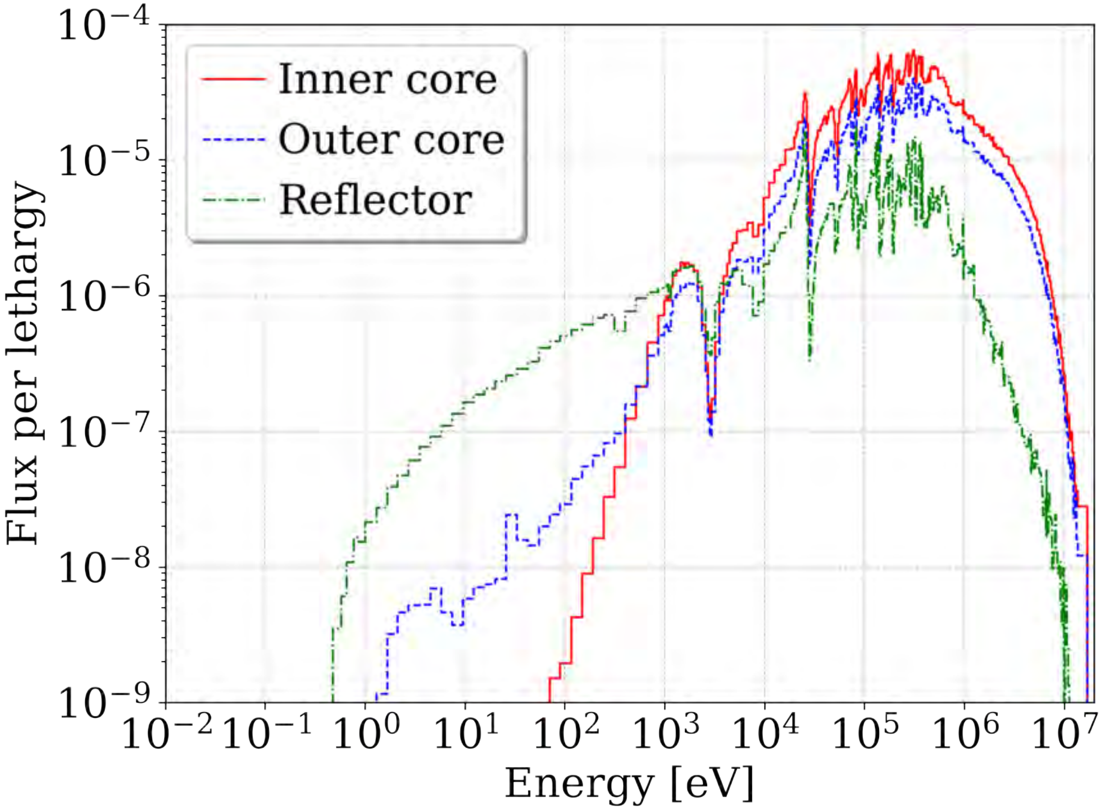
SCALE Model



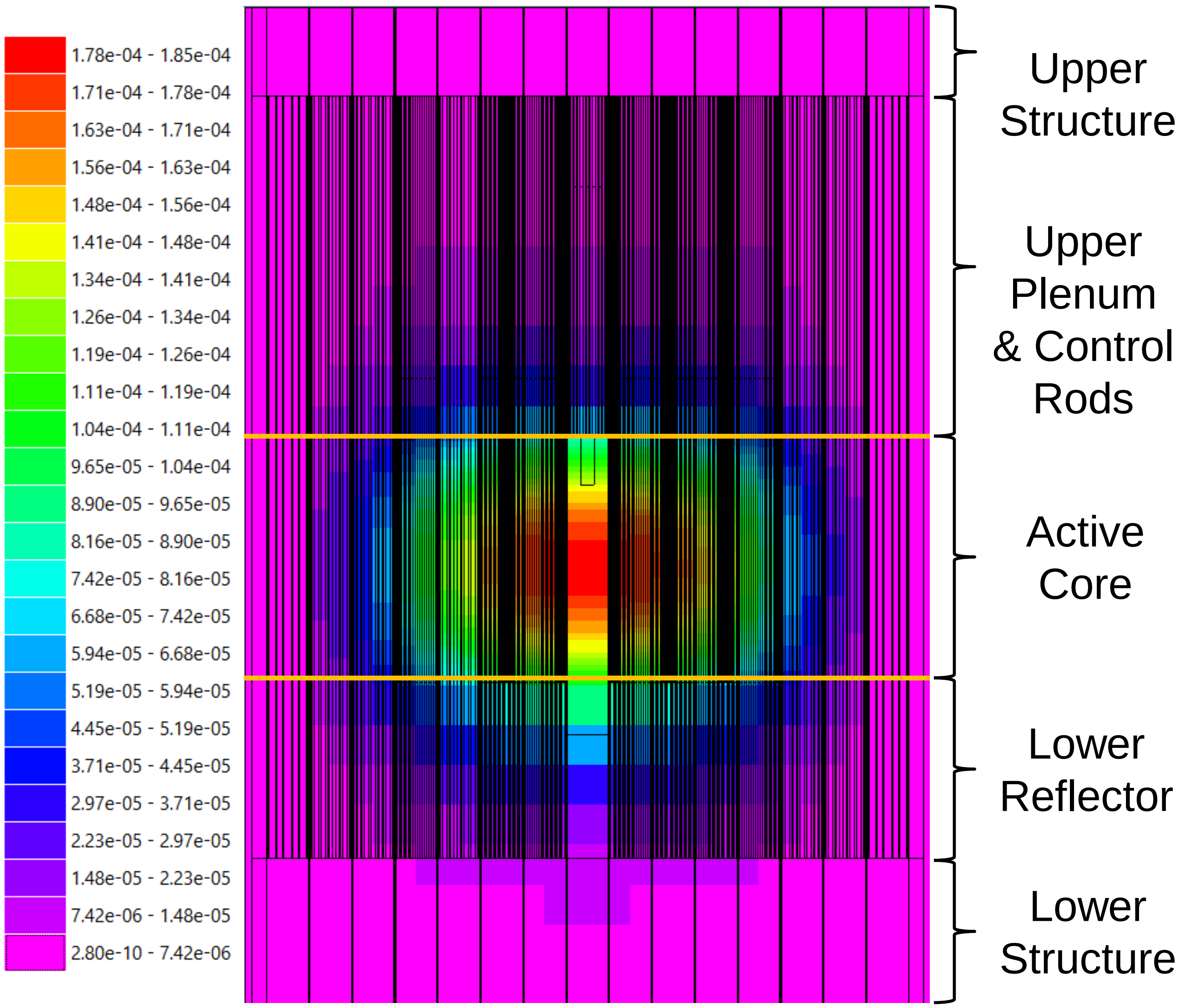
ABTR Core [2]

Note: The displayed SCALE model does not display coolant to avoid confusion with the withdrawn control assemblies.

Neutron Flux in the BOEC Core



Energy-dependent Flux Spectrum



Total Flux (Linear Scale)

Note: The displayed flux is the flux per fission neutron divided by the mesh voxel volume.

Verification of BOEC SCALE model

- Verification of the BOEC* SCALE model was performed relative to:
  - ANL ABTR reference design description [1]
  - INL ABTR Multiphysics report [3]

Code	XS Library	k_{eff}	Difference (pcm)
Serpent [3]	ENDF/B-VII.1	1.03055 ± 0.00002	(ref)
SCALE	ENDF/B-VII.1	1.03019 ± 0.00004	37±4
	ENDF/B-VIII.0	1.03152 ± 0.00004	-97±4

Code	XS Library	β_{eff}
DIF3D [1]	ENDF/B-V.2	0.00330
Serpent [3]	ENDF/B-VII.1	0.00330
SCALE	ENDF/B-VII.1	0.00331 ± 0.00005

**EOEC values not available for verification*

[1] Y. I. Chang, et al. (2006). Advanced Burner Test Reactor Preconceptual Design Report, Technical Report ANL-AFCI 173, Argonne National Laboratory, Argonne, IL.
[3] C. M. Mueller, et al. (2021). NRC Multiphysics Analysis Capability Deployment FY2021 – Part 2, Technical Report INL/EXT-21-62522, Idaho National Laboratory, Idaho Falls, ID.

Reactivity Effects

Reactivity Coefficients

- Litany of model perturbations were performed to calculate reactivity coefficients
- Axial Fuel Expansion:
  - A 1% expansion was considered, representing a 575K increase in fuel temperature
  - Density was correspondingly adjusted
- Radial Grid Plate Expansion:
  - Uniform, radial thermal expansion of the SS-316 grid plate (increasing assembly pitch)
  - Cold (293K) to operating (628K)
  - Pitch increase of 0.087 cm (0.6%)

Feedback Effect	SCALE
Axial Fuel Expansion Coefficient (cents/K)	-0.135 ± 0.003
Radial Grid Plate Expansion Coefficient (cents/K)	-0.338 ± 0.007

Reactivity Coefficients, cont.

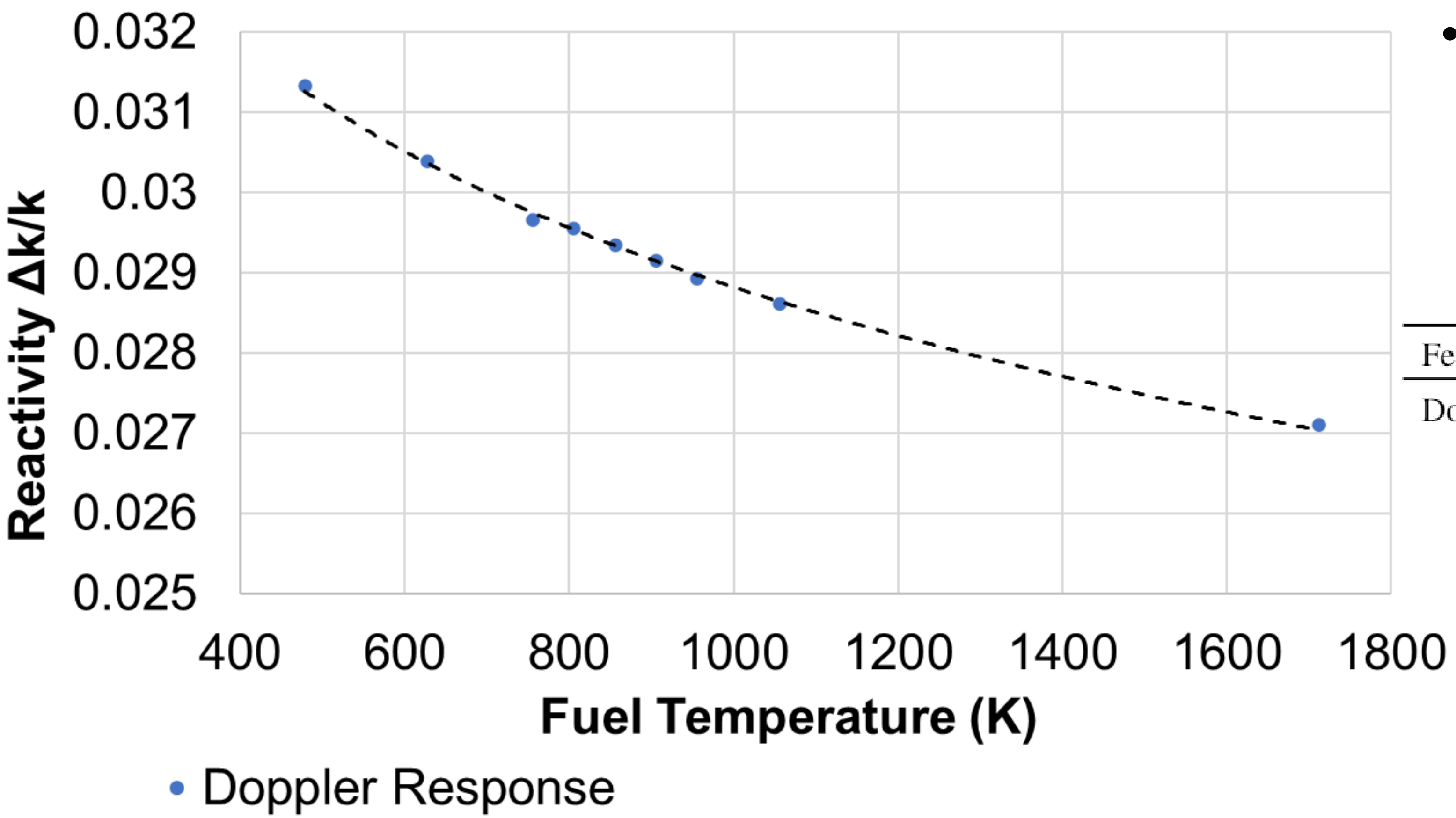
- Fuel Density:
  - A 1% density reduction while conserving dimensions (decreasing mass)
  - Enhanced response relative to axial fuel expansion due to lost mass
- Structure Density:
  - All HT-9 components (cladding, ducts, reflector, structure, followers, barrel)
  - A 1% density reduction results from a 720K increase (decreasing mass)
- Sodium Void Worth:
  - Flowing sodium was voided within fuel assembly ducts, active fuel region and above
  - Varied from literature values, but known issues exist in calculating void worth with homogenized methods common for SFRs, as well as an XS library dependence [4,5]

Feedback Effect	SCALE
Fuel Density Coefficient (cents/K)	-0.244 ± 0.004
Structure Density Coefficient (cents/K)	-0.013 ± 0.002
Sodium Void Worth ($)	-0.462 ± 0.016

[4] W. S. Yang, et al. (2007).Preliminary Validation Studies of Existing Neutronics Analysis Tools for Advanced Burner Reactor Design Applications Technical Report ANL-AFCI-186, Argonne National Laboratory.
[5] NEA (2016).Benchmark for Neutronic Analysis of Sodium-cooled Fast Reactor Cores with Various Fuel Types and Core Sizes Technical Report NEA/NSC/R(2015)9, Nuclear Energy Agency.

Reactivity Coefficients, cont.

- Doppler:
  - Nine fuel temperatures were utilized to determine the Doppler coefficient
  - Logarithmic response expected from fast spectrum HPR experience, so coefficient is calculated as derivative at nominal fuel temperature (with respect to reactivity, not k_{eff})

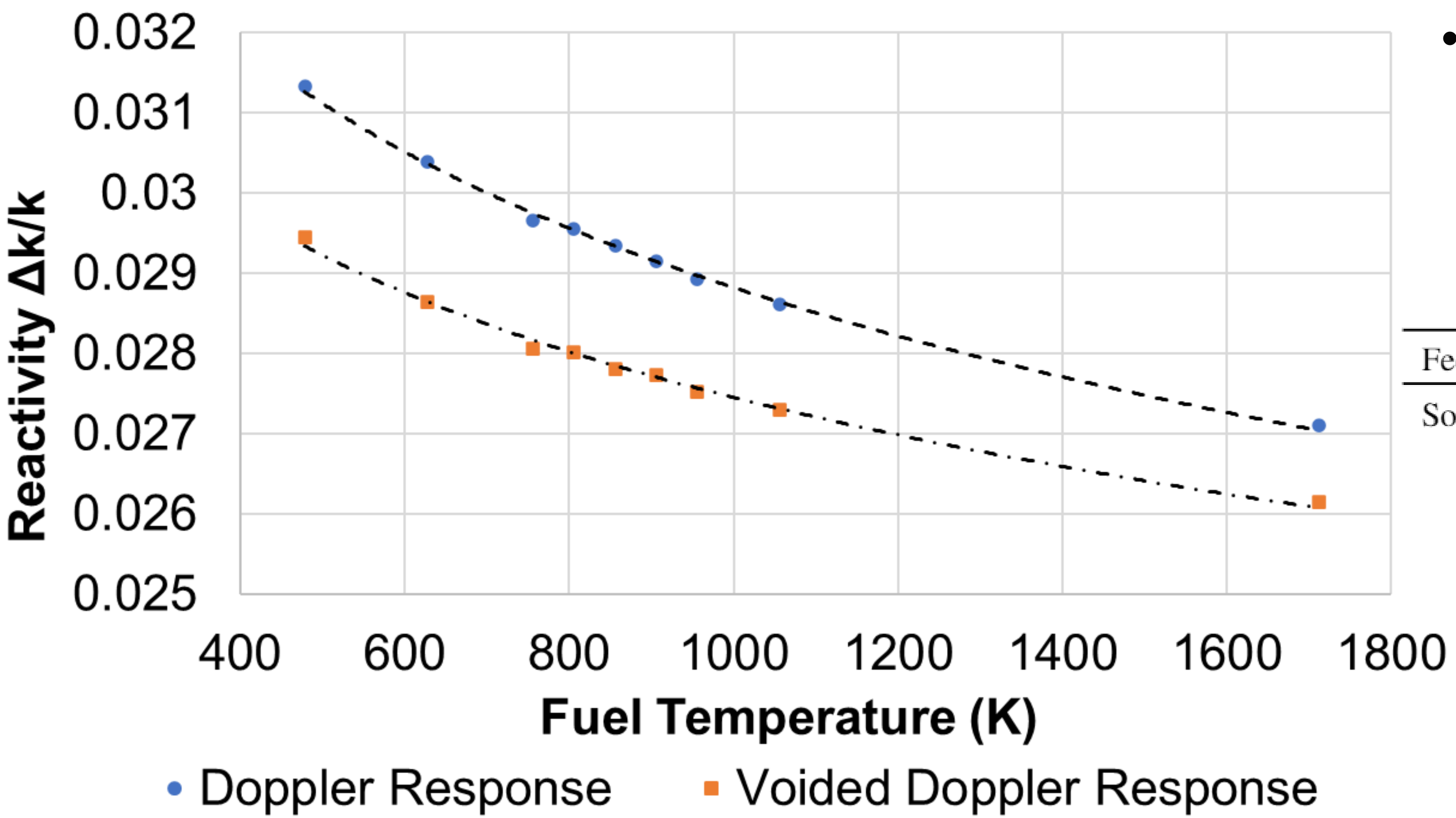


- Linear approach can cause underestimation of Doppler coefficient
  - -0.079 cents/K linear with 2 points
  - -0.098 cents/K linear with 9 points

Feedback Effect	SCALE
Doppler Coefficient (cents/K)	−0.117 ± 0.003

Reactivity Coefficients, cont.

- Sodium Voided Doppler:
  - Nine fuel temperatures were utilized to determine the Doppler coefficient
  - Logarithmic response expected from fast spectrum HPR experience, so coefficient is calculated as derivative at nominal fuel temperature (with respect to reactivity, not k_{eff})



- Linear approach can cause underestimation of Doppler coefficient
  - -0.059 cents/K linear with 2 points
  - -0.075 cents/K linear with 9 points

Feedback Effect	SCALE
Sodium Voided Doppler Coefficient (cents/K)	−0.090 ± 0.003

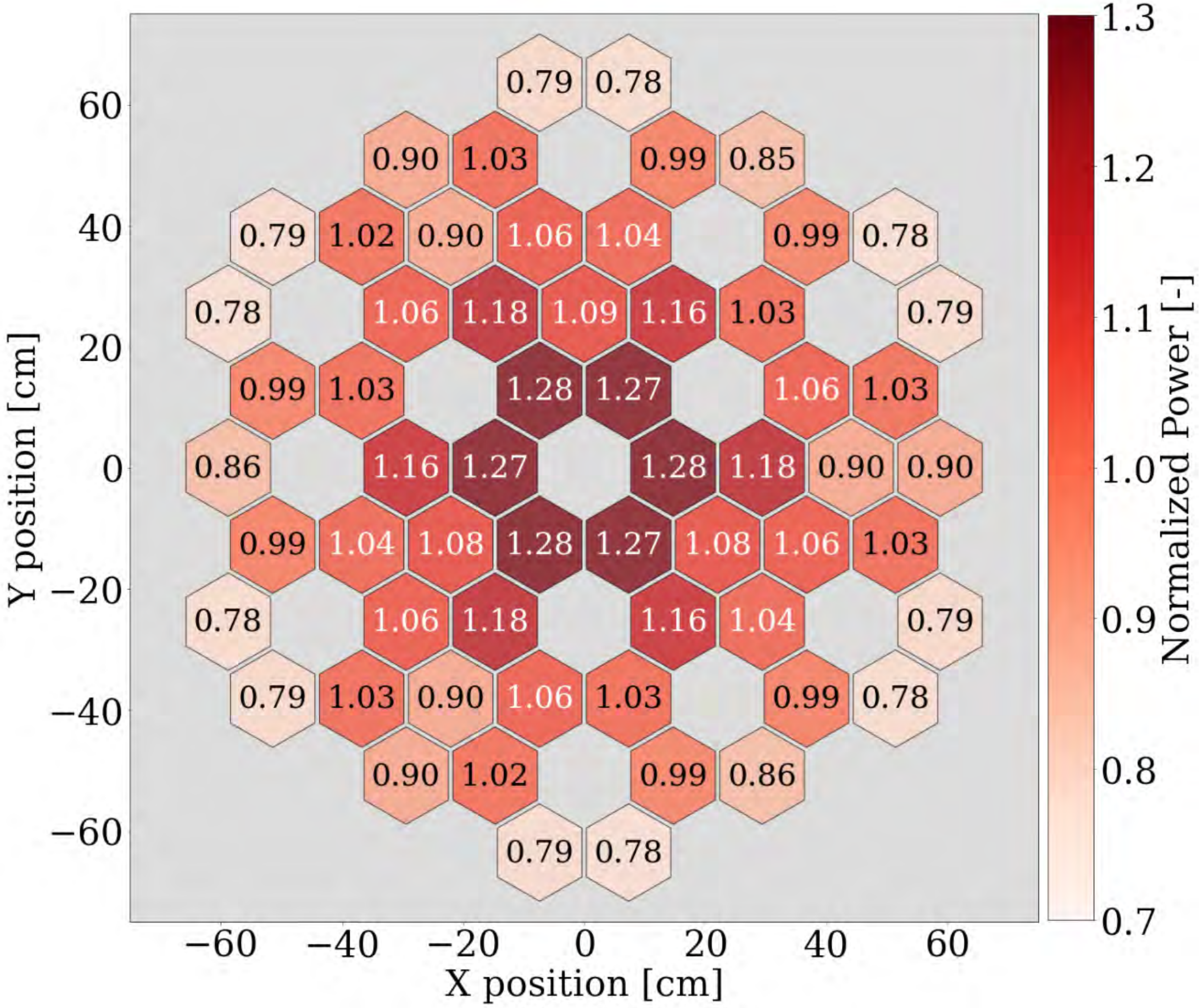
Fuel Depletion

TRITON Modeling

- CSAS-KENO input was converted to TRITON-KENO input for depletion
- CSAS to TRITON conversion involves:
  - Fuel region discretization
    - Individual assembly definitions (60 fuel assemblies) for radial discretization
    - 10 axial zones per assembly for axial discretization
    - 600 total depletion zones for power profiling and tracking inventory
  - Applying specific power for the system
    - 61.7 MW/MTHM
  - Determining the appropriate number of depletion steps and spacing for accurate flux response evolution while maintaining computational efficiency
    - 6 burnup points over the 4-month cycle
  - Nuclide tracking between depletion steps
    - 95 relevant fission products and actinides
  - Depleting materials of interest (fuel)
    - All 600 discretized depletion zones

TRITON Modeling

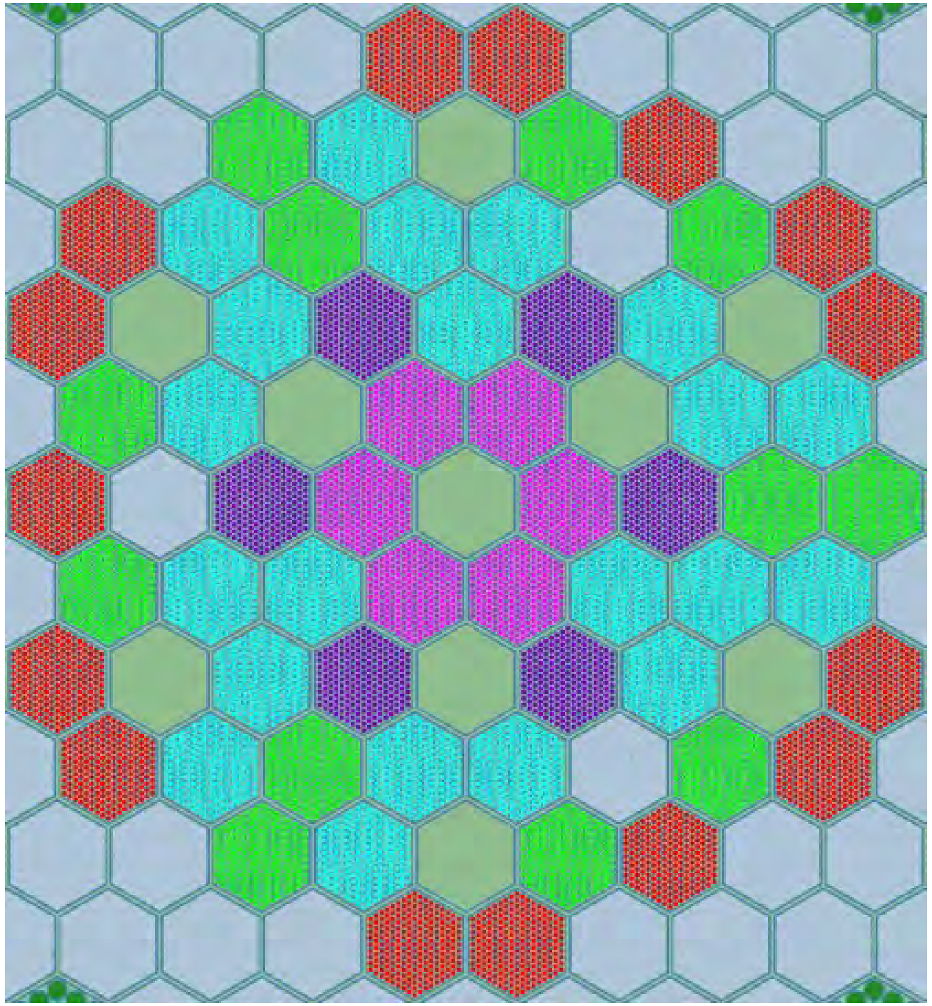
- Analysis of normalized axially integrated assembly power distribution informed grouping of assemblies:
  - Group 1 (0.7-0.9)
  - Group 2 (0.9-1.0)
  - Group 3 (1.0-1.1)
  - Group 4 (1.1-1.2)
  - Group 5 (1.2-1.3)
- Grouping allows for simpler data transfer to MELCOR (5 radial groups, 10 axial zones) vs pointwise (600 depletion zones)



Power Map

TRITON Modeling

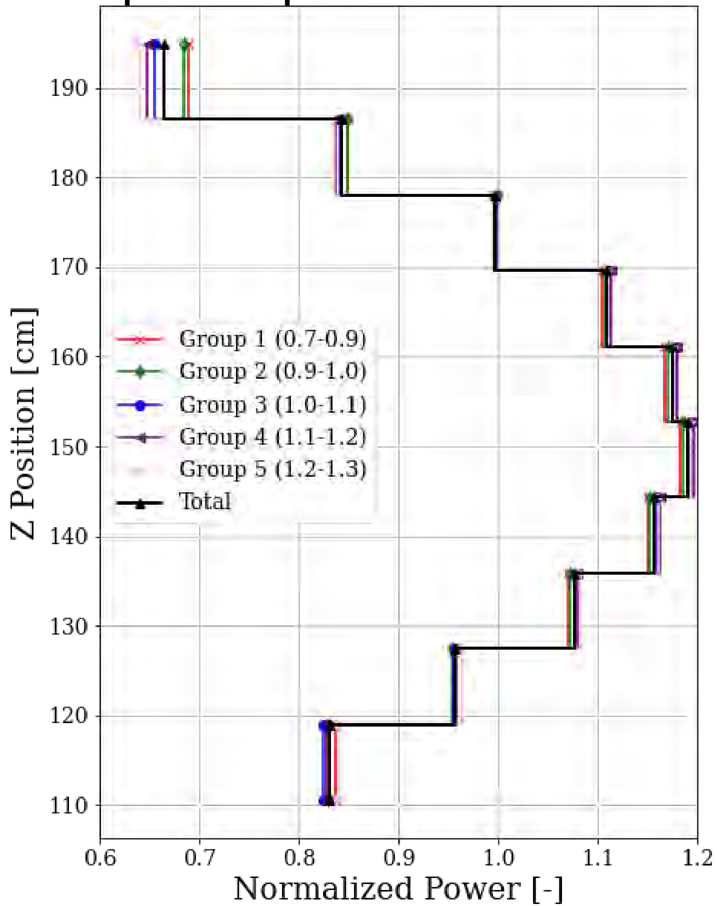
- Analysis of normalized axially integrated assembly power distribution informed grouping of assemblies:
  - Group 1 (0.7-0.9)
  - Group 2 (0.9-1.0)
  - Group 3 (1.0-1.1)
  - Group 4 (1.1-1.2)
  - Group 5 (1.2-1.3)
- Grouping allows for simpler data transfer to MELCOR (5 radial groups, 10 axial zones) vs pointwise (600 depletion zones)

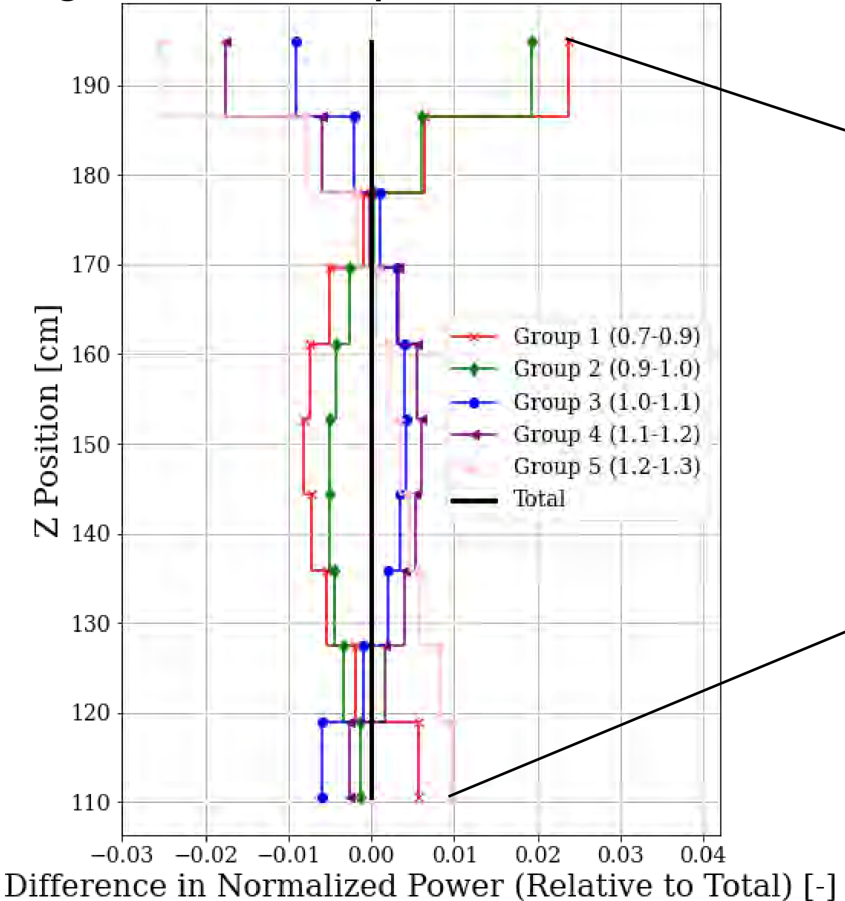


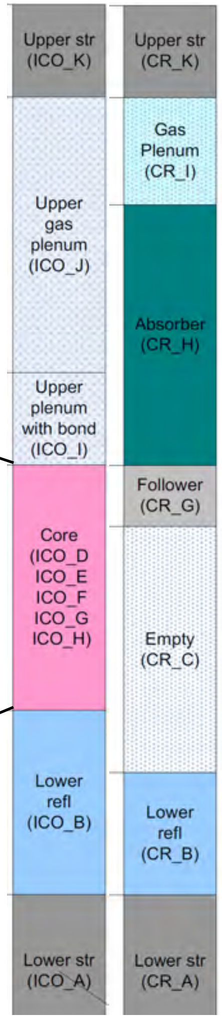
ABTR Model with Color-Coded Assemblies

Power Distribution

- Axial profile steady radially throughout the core
- Upper regions are slightly more variable and lower power with control assemblies withdrawn and a lack of upper reflector
- Axial profile provided as the resulting normalized power from all assemblies (**Total**)



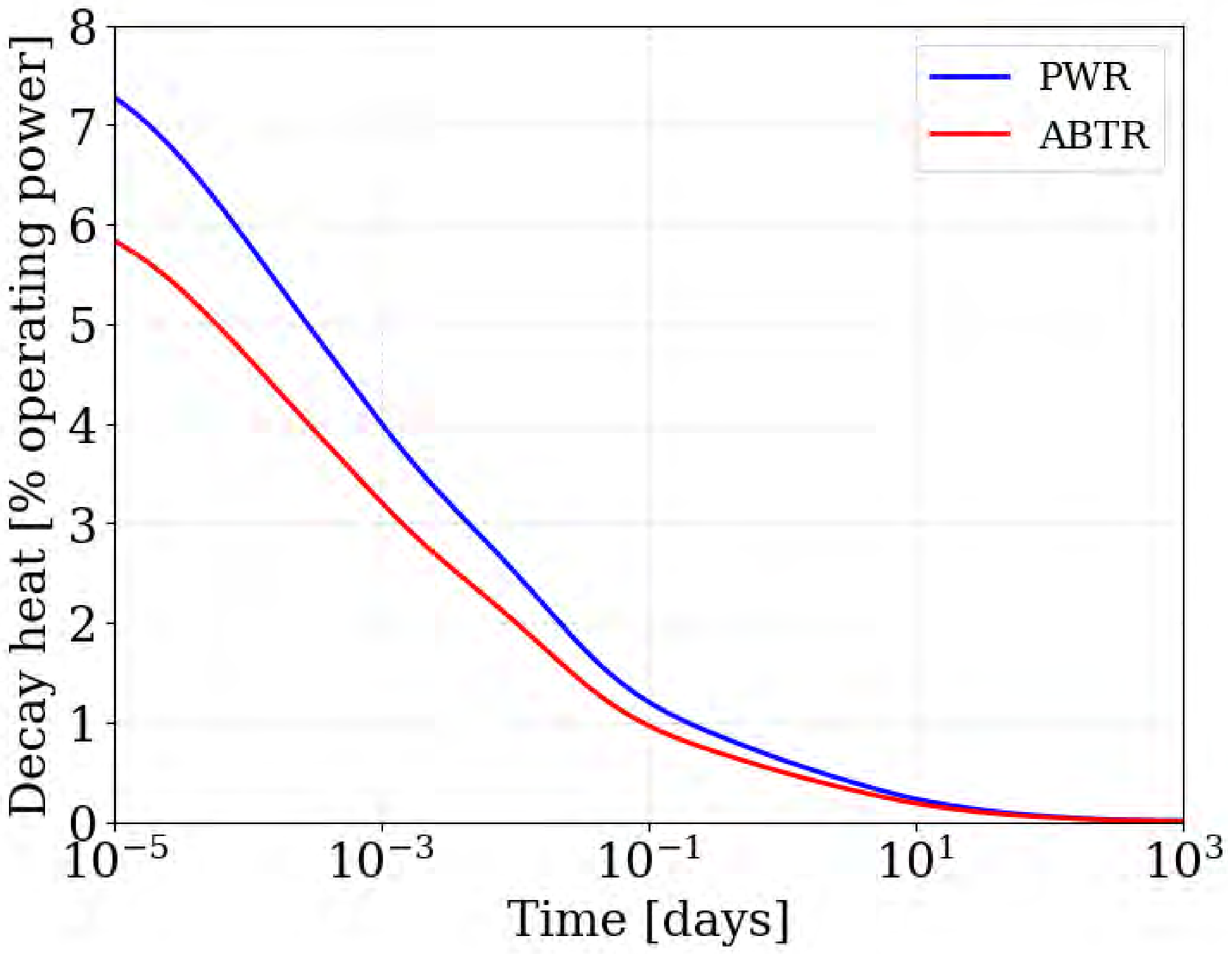




Fuel and Control Assembly Cross Section [2]

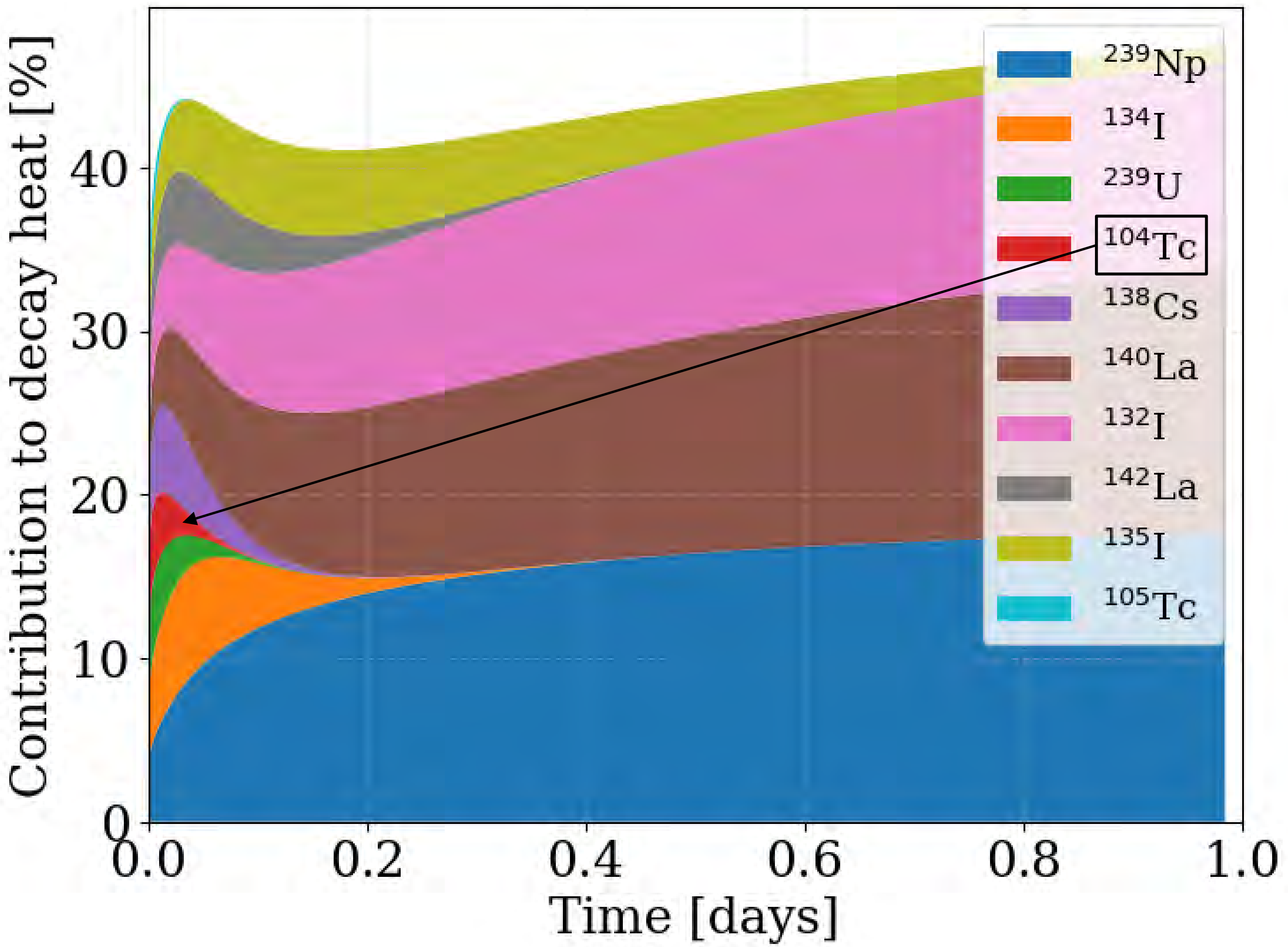
EOEC Inventories and Decay Heat

- A full-core, explicit assembly TRITON model was used to deplete from BOEC→EOEC, generating power and nuclide inventory distributions
- Nuclide inventories are available for 600 depletion zones at 6 time points over the 4-month cycle
- Information flow to MELCOR
  - OBIWAN utility from SCALE 6.3 converts ORIGEN binary concentration files into Inventory Interface JSON files (ii.json)
  - Python script converts ii.json to a MELCOR DCH input file (mass and decay heat by element group)



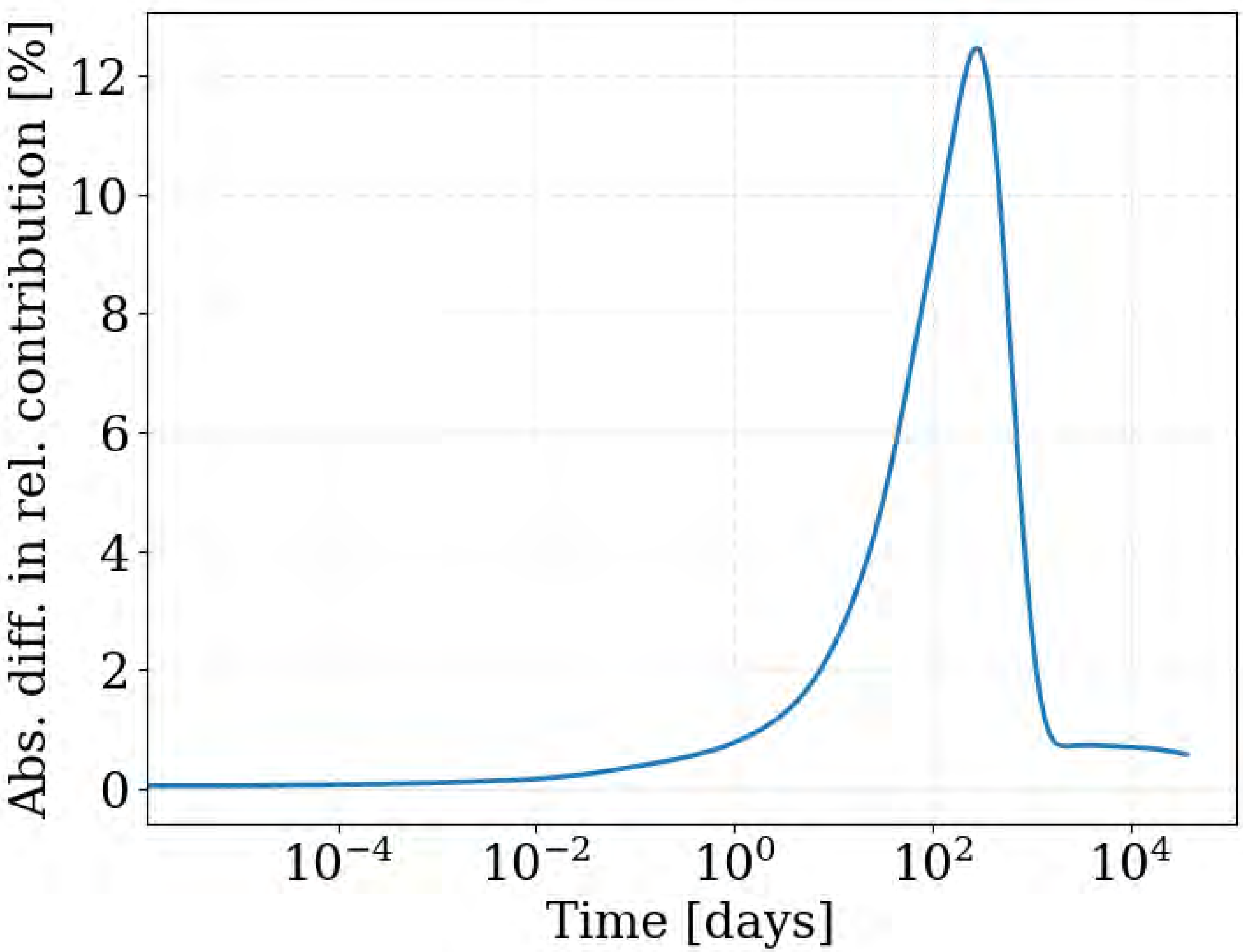
Total decay heat after shutdown

Core Decay Heat after Shutdown



- Top 10 decay heat-producing isotopes in the first day following shutdown
- Inventory consistent with other reactor designs, except Tc-104 ($T_{1/2}$ =18 min)
  - Tc-104 is a top 10 contributor to decay heat in the first 30 seconds (2.3%) and 30 days (3%)
  - Fission yield of Tc-104 ~10x higher for Pu-239 vs. U-235
  - Pu content of initial core is much higher than other designs
  - Notable for the difference from other designs—not magnitude

Core Decay Heat after Shutdown, cont.



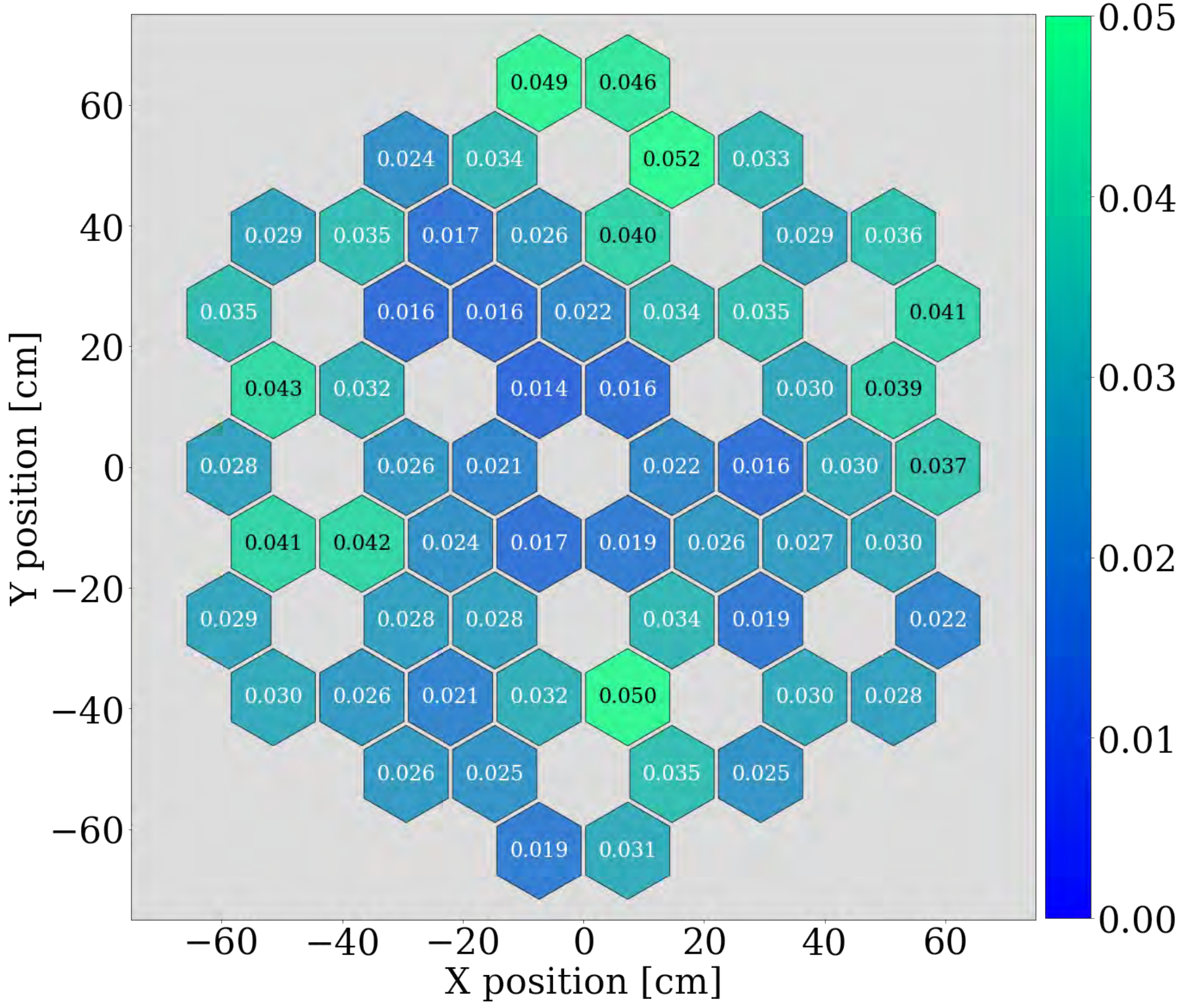
Difference in Cm-242 contribution to decay heat between ABTR and PWR

- Inventory in the first 30 days consistent with other reactor designs, except Cm-242
- ~12% additional decay heat at ~100 days due Cm-242
  - Initial loading contains higher trans-uranic (TRU) concentrations
  - Cm-242 generated through Am-242 in activation chains

Additional Studies in Support of MELCOR Analyses

Statistical Convergence of Power Distribution

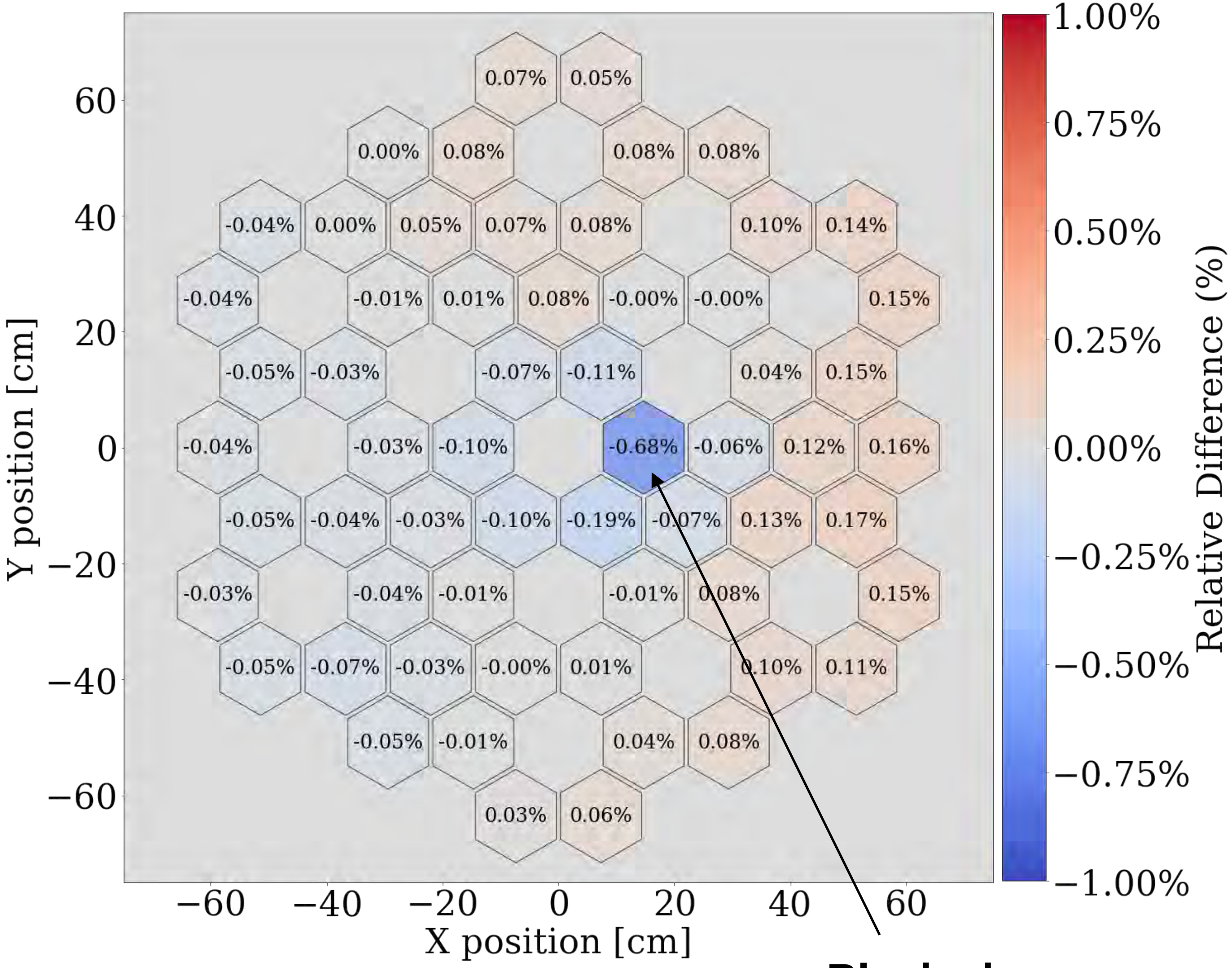
- In a Monte Carlo simulation, results have statistical errors
- Random number seed variations allow an estimate of the average power and the corresponding statistical error
- Estimating the error used here to confirm convergence
  - Max. error of 0.1%, average of 0.05%
  - Understanding the magnitude of the statistical error allows to distinguish impact of actual power perturbations from statistical noise



Statistical error (%) in assembly power

Single Assembly Sodium Voiding Effect on Power Distribution

- MELCOR scenario considers single assembly blockage
  - Specifics of scenario to be detailed by the MELCOR team
- Effect of single assembly voiding was investigated to confirm that the provided nominal power profile is applicable
- Comparison of power maps shows that most differences are at the level of statistical noise (<0.1%)
- **Blocked assembly shows 0.6% difference in power**



Blocked assembly

Additional Worth Estimates

- Control assembly worths were calculated at BOEC by calculating reactivity differences with insertion
  - Each bank is individually sufficient for subcriticality
  - Demonstrated agreement with the design report [1]

	Inserted Control Assemblies	DIF3D Worth($) [1]	SCALE Worth($)
Primary Bank			
-Central Assembly	1	6.53	6.49 ± 0.10
-5th Row	6	15.95	15.33 ± 0.23
-All Primary	7	23.52	22.07 ± 0.33
Secondary Bank			
-All Secondary	3	16.38	15.77 ± 0.23

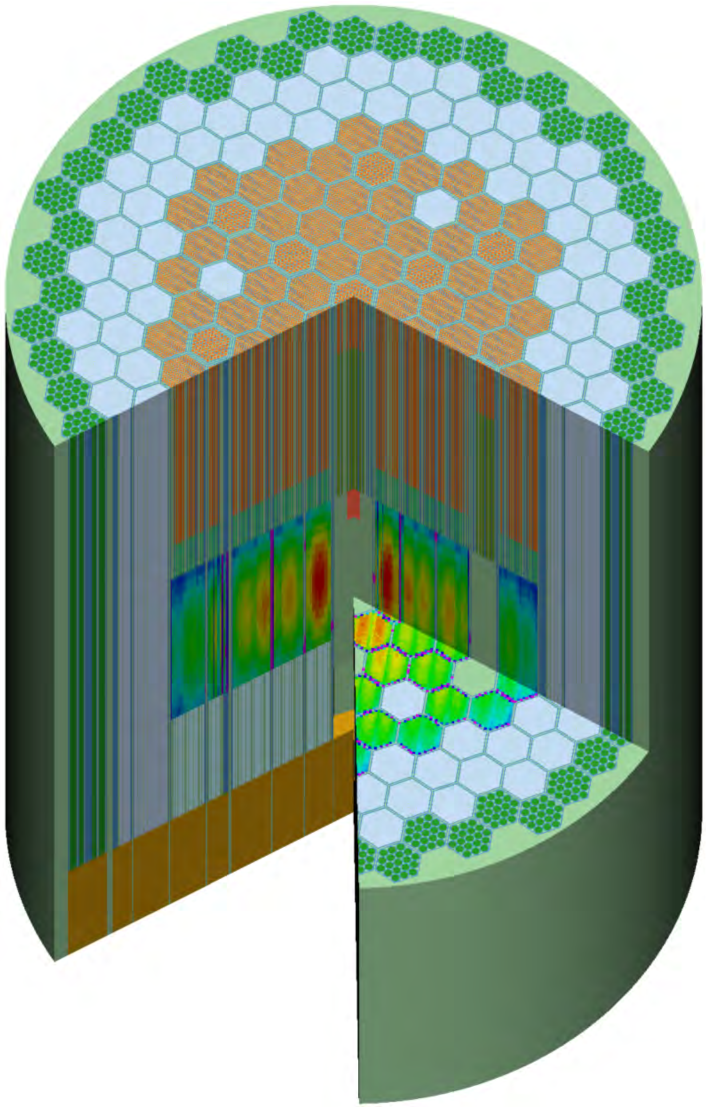
- Xe-135 worth: 9 ± 5 pcm (confirmed negligible for ABTR)

[1] Y. I. Chang, et al. (2006). Advanced Burner Test Reactor Preconceptual Design Report, Technical Report ANL-AFCI 173, Argonne National Laboratory, Argonne, IL.

Summary and Conclusions

SCALE SFR Summary

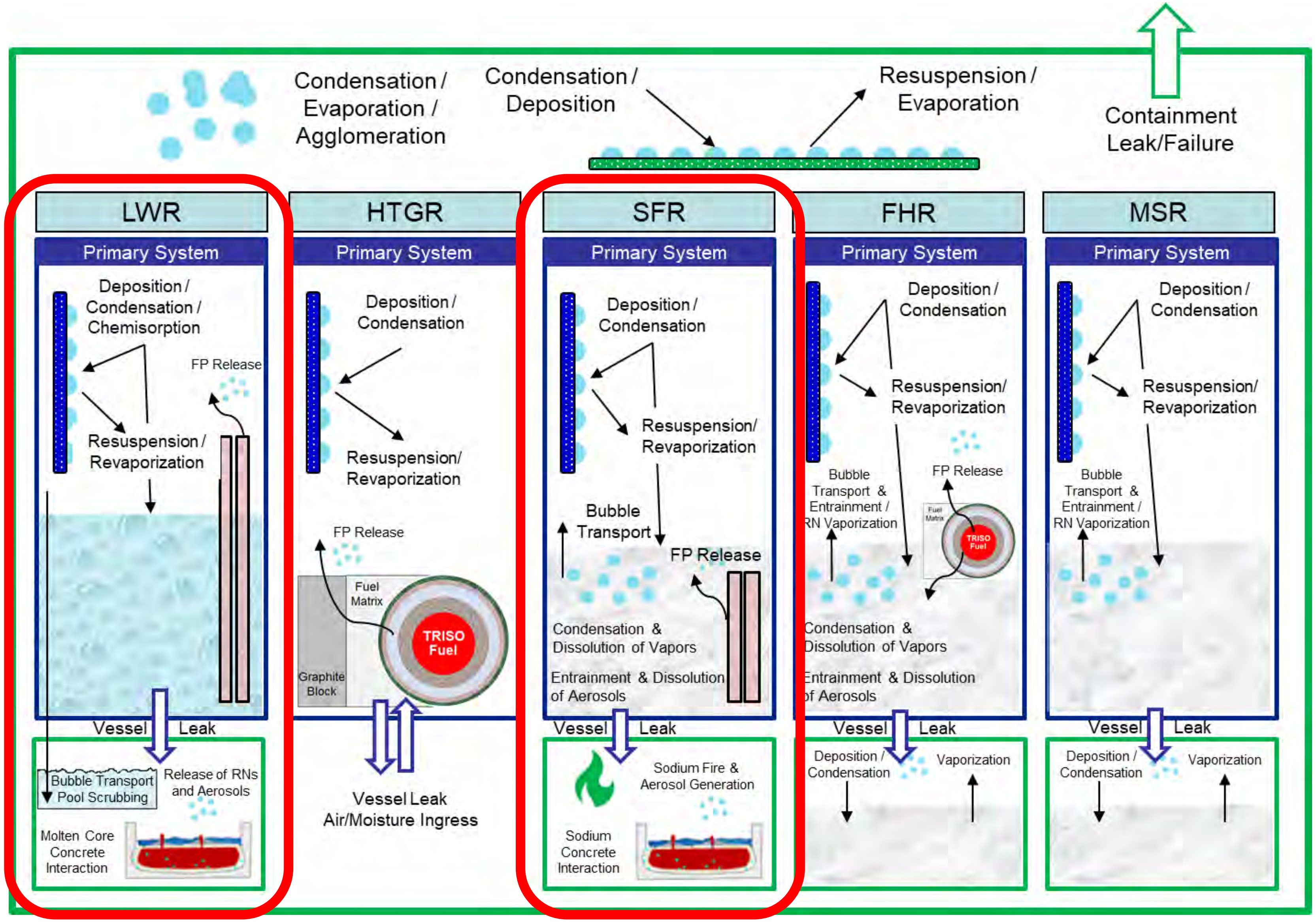
- Fast-spectrum SFR modeling with SCALE
  - Continuous energy Monte Carlo neutronics with KENO and ORIGEN for depletion are high-fidelity, system-independent
  - Consistency with code-to-code comparisons in all verification studies
- Key results
  - Reactivity Coefficients (including non-linear Doppler)
  - Full 3D power distributions (axial profile is sufficient for MELCOR)
  - Inventory and Decay Heat
    - Cm-242 and Tc-104 have notable (but small) differences in SFRs utilizing U/TRU-Zr10% compared to PWRs
- Future SFR work:
  - Additional reactivity analyses for further insights into SFR behavior
  - Analyses of scenarios in the SFR fuel cycle (Volume 5)



3D SCALE ABTR Core with Fission Density Overlay

MELCOR Sodium Fast Reactor Models

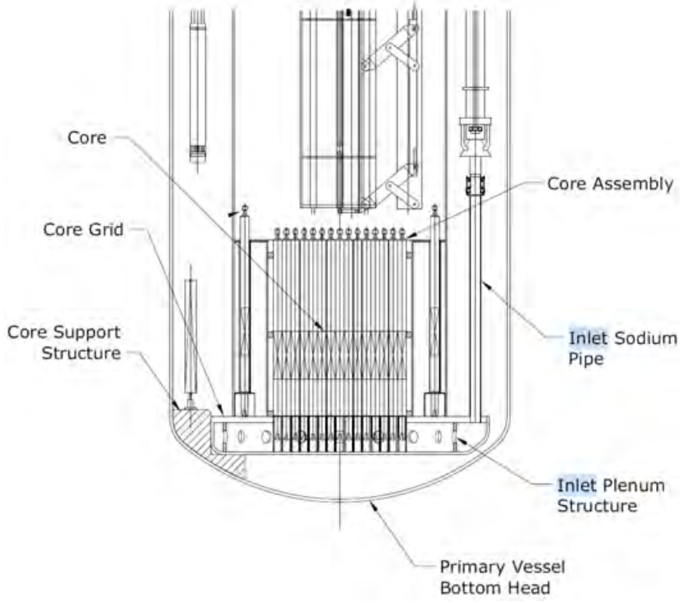
Evolution of SFR Modeling

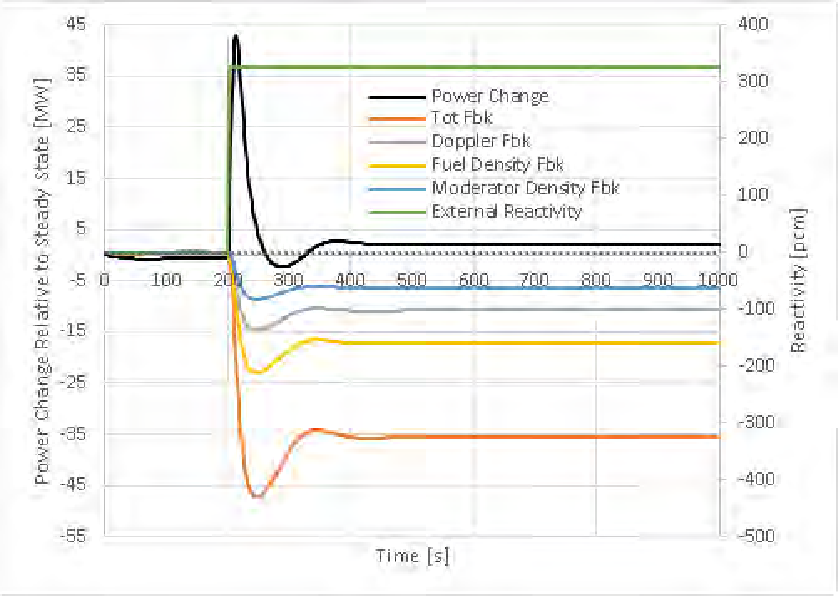


Modeling SFR Accidents with MELCOR

MELCOR SFR Modeling

- SFR materials
  - U-10Zr metallic fuel, HT-9 cladding, and sodium bond
  - Sodium fluid EOS
- Fast reactor point kinetics
- Establishing initial conditions
  - Decay heat, radionuclide inventory, and power distribution specification (SCALE)
  - Initial fission product gas distribution (gas plenum, closed and open pores)
  - Fuel expansion and swelling geometry
- Core damage progression
  - Fuel melting
  - Clad pressure boundary failure, melting and candling
  - Degraded fuel region molten and particulate debris behavior
- Radionuclide release and transport
  - Gap and plenum release
  - Molten fuel fission gas release
  - Thermal release models





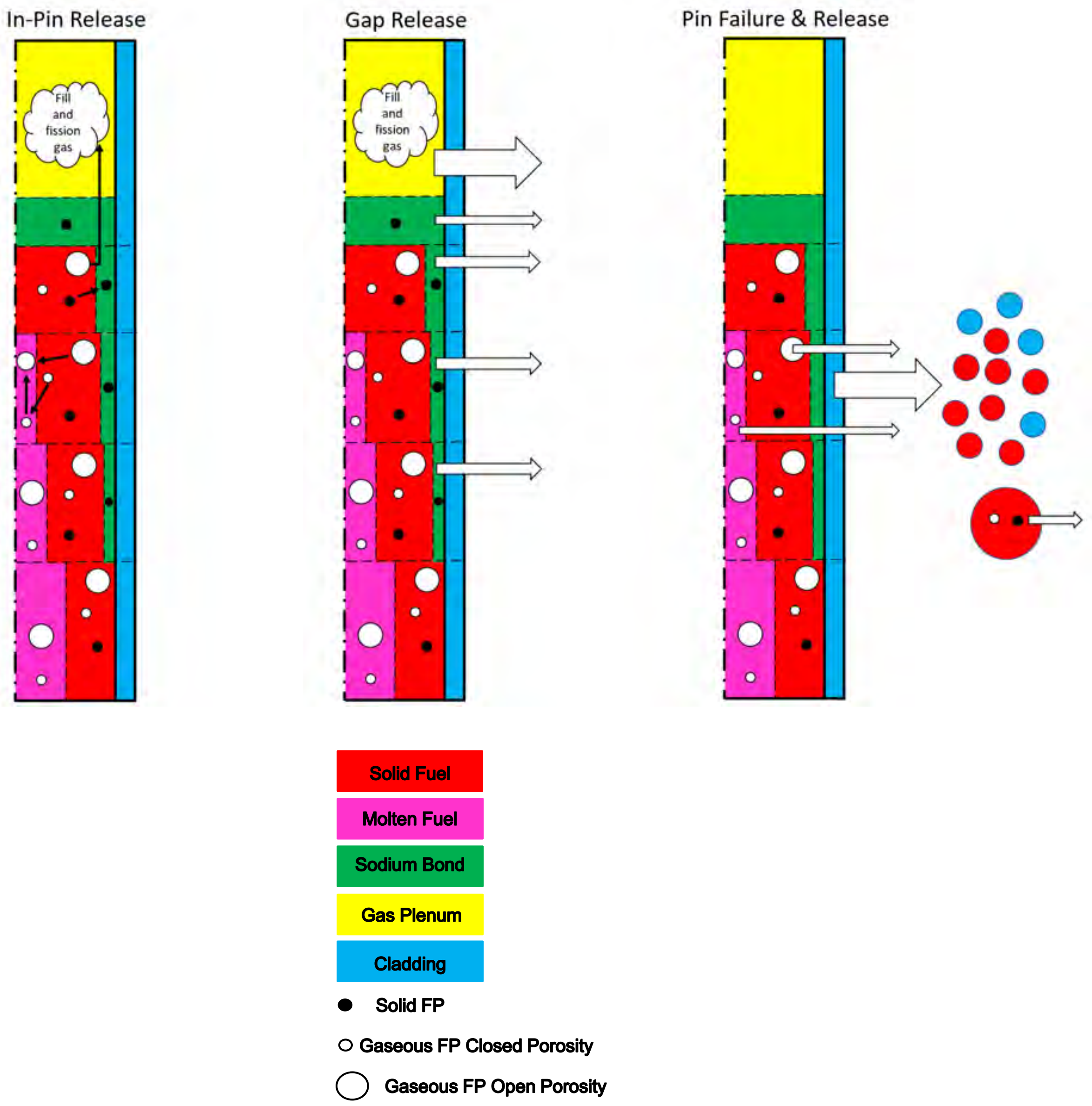
Fuel damage progression and radionuclide release

Models added to simulate unique metal fuel behavior

- Fuel melting prior to cladding failure
- Evolution of closed pores to interconnected, open pores
- Existing models of candling, molten pools, particulate debris

Fission product release characterized by distinct phases

- In-pin release - migration of fission products to fission product plenum and sodium bond
- Gap release – burst release of plenum gases and fission products in the bond
- Pin failure & release – radionuclide releases from hot fuel debris



Sodium Fire Modeling

Atmospheric chemistry + aerosol generation

- Implementation and validation of MELCOR
  - o Spray model is based on NACOM spray model from BNL
  - o Pool fire model is based on SOFIRE-II code from ANL
- Ongoing benchmarks with JAEA F7 pool and spray fire experiments
- Previous benchmarks to ABCOVE AB5 and AB1 tests

(A1) $Na(l) + H_2O\,(l) \rightarrow NaOH(a) + \frac{1}{2}H_2$

(A2) $2\,Na(g,l) + H_2O\,(g,l) \rightarrow Na_2O(a) + H_2$

(A3) $2\,Na(g,l,a) + \frac{1}{2}O_2$ or $O_2 \rightarrow Na_2O(a)$ or $Na_2O_2(a)$

(A4) $Na_2O_2(a) + 2\,Na(g,l) \rightarrow 2\,Na_2O(a)$

(A5)

- $Na_2O(a) + H_2O\,(g,l) \rightarrow 2NaOH(a)$
- $Na_2O_2(a) + H_2O\,(g,l) \rightarrow 2NaOH(a) + 0.5O_2$

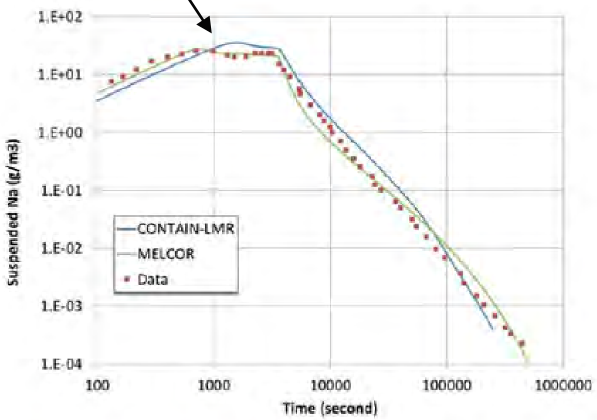


Figure 33. Suspended Na Aerosol Mass - AB1

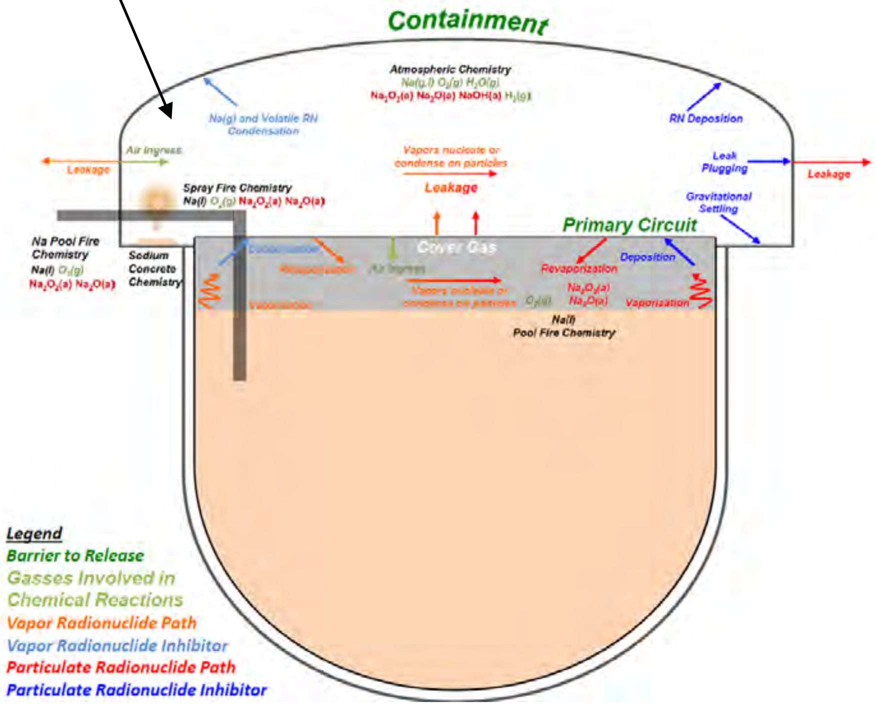


Figure adapted from ANL-ART-3

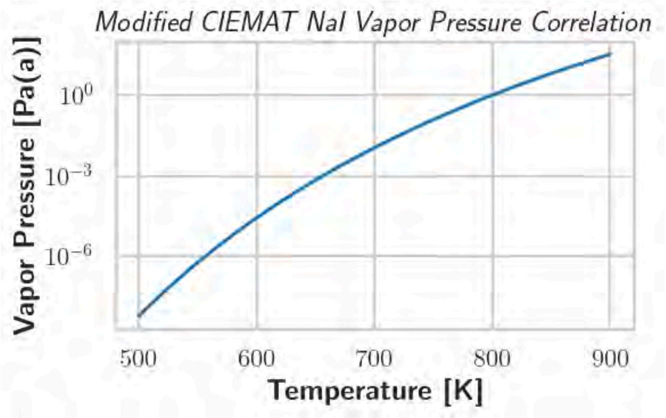
GRTR – Generalized Radionuclide Transport and Retention

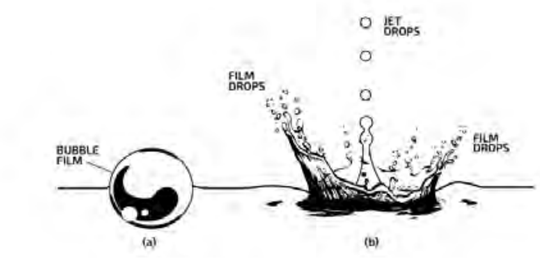
Tracks fission products and determines how much is released from liquid to atmosphere

Characterizes evolution of fission products between different physico-chemical forms

GRTR mass transport modeling essential for understanding effect of sodium on source term

- Retention in sodium of many important radionuclides as a function of solubility and vapor pressure
- Bubble transport and bursting
- Deposition on structural surfaces in sodium pool and core
- Jet breakup and splashing





GRTR and Integral MELCOR Simulations

Inputs to GRTR Model

- Radionuclide mass in (or released to) liquid pool
- Chemical speciation
- Pressure in hydrodynamic volume
- Temperature in *regions* of hydrodynamic volume (e.g., liquid and atmosphere)
- Advective flows of liquid and atmosphere between hydrodynamic volumes

GRTR Physico-Chemical Transport Dynamics

- Soluble radionuclide form mass
- Colloidal radionuclide form mass
- Deposited radionuclide form mass
- Gaseous radionuclide mass

Advective and Fission/Transmutation Dynamics

- Advection of radionuclides in liquid pool or atmosphere
- Decay of radionuclides in hydrodynamic control volume
  Coupling with ORIGEN

MELCOR SFR Plant Model and Source Term Analysis

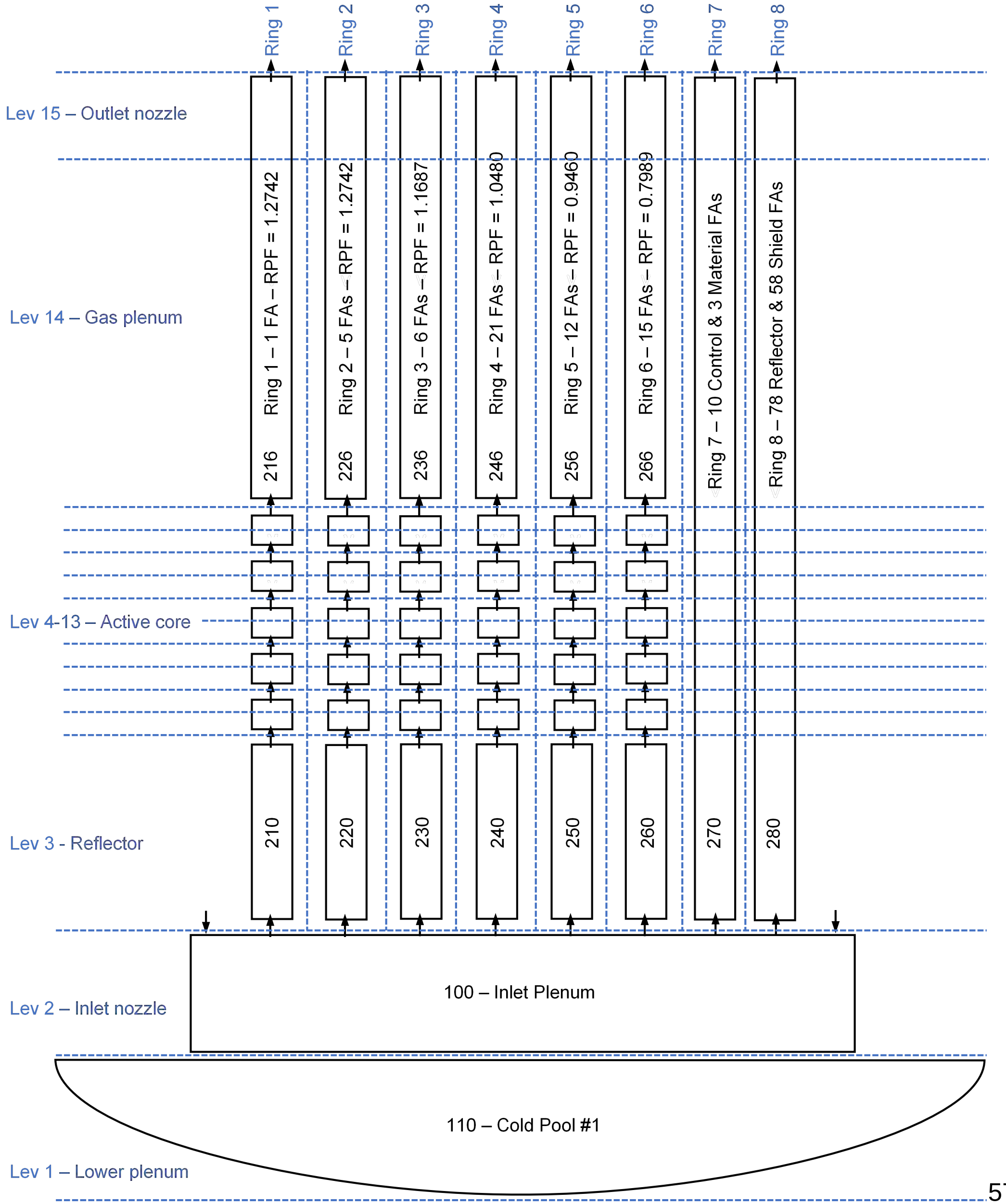
Core

Core nodalization – light blue lines

- Subdivided into 15 axial levels and 8 radial rings
- Core divided according to assembly power and function (similar to SFP modeling)
  - Ring 1 through 6 = 60 fueled assemblies combined according to power
  - Ring 7 = 10 control and 3 material test assemblies
  - Ring 8 = 78 reflector and 58 shield assemblies
  - The 8 rings share a common inlet plenum and the lower cold pool

Fluid flow nodalization – black boxes

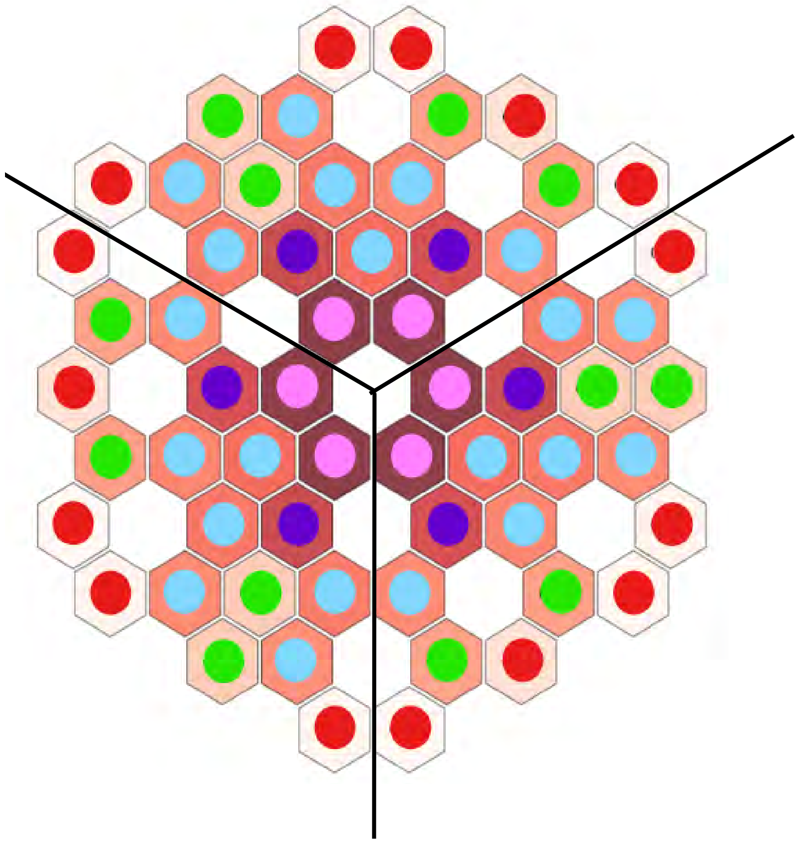
- Sodium enters through the inlet plenum and flows into the assemblies



MELCOR core region mapping to SCALE

SCALE Radial Zones

1
2
3
4
5



MELCOR Ring 8 (78 reflector and 58 shield assemblies)

MELCOR Ring 7 (10 control and 3 material test assemblies)

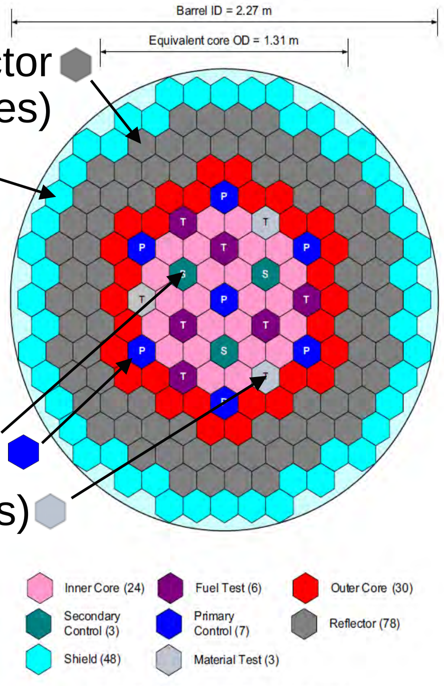


Figure II.1-1 Reference ABTR Core Configuration

MELCOR radial mapping to SCALE

MELCOR axial mapping

SCALE Radial Zone (r)	1	2	3	4	5	
MELCOR Radial Zone (r)	6	5	4	3	2	1
Number of Assemblies	15	12	21	6	5	1
Assembly Power Factor	0.80	0.95	1.05	1.17	1.27	

- 1 SCALE level per MELCOR COR level
- 2 SCALE levels per MELCOR CVH level

Vessel

All primary system sodium is contained within the vessel

Sodium exits into a hot pool and circulates through the shell side of 2 intermediate heat exchangers (iHX)

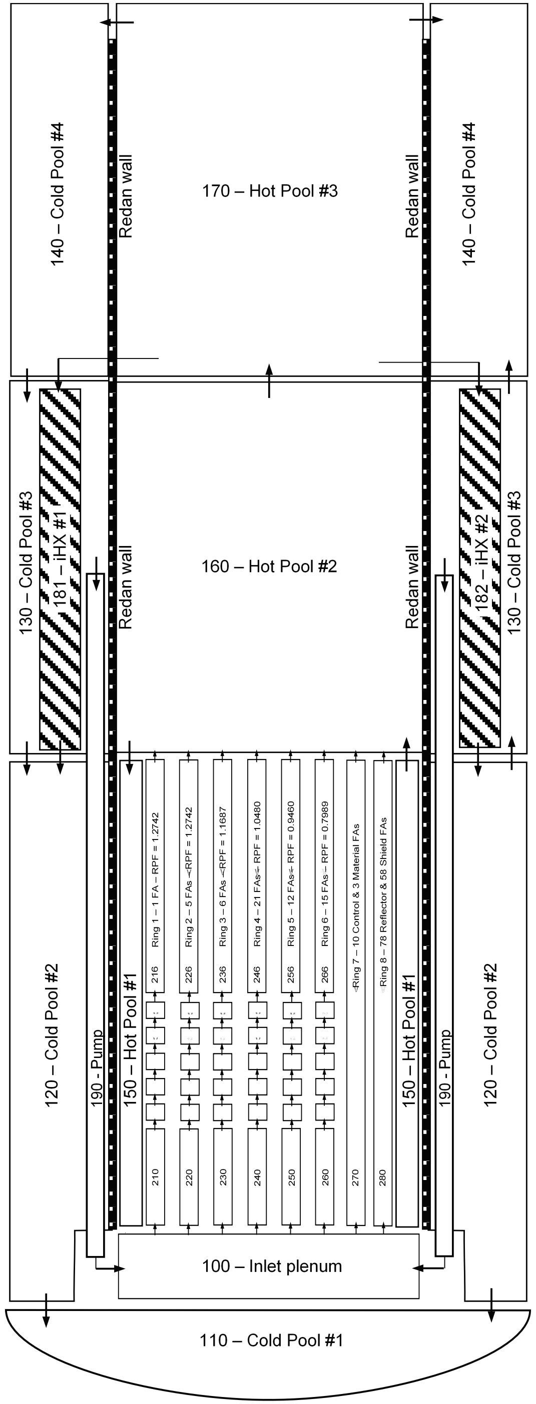
A redan (wall) separates the hot pool from the cold pool

2 EM or mechanical pumps circulate sodium into the vessel inlet

Free surfaces at the top of the hot and cold pools

Argon gas above the free surfaces with connection to the cover-gas system

- Assumed leak path for fission products



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

Direct Reactor Auxiliary Cooling System (DRACS)

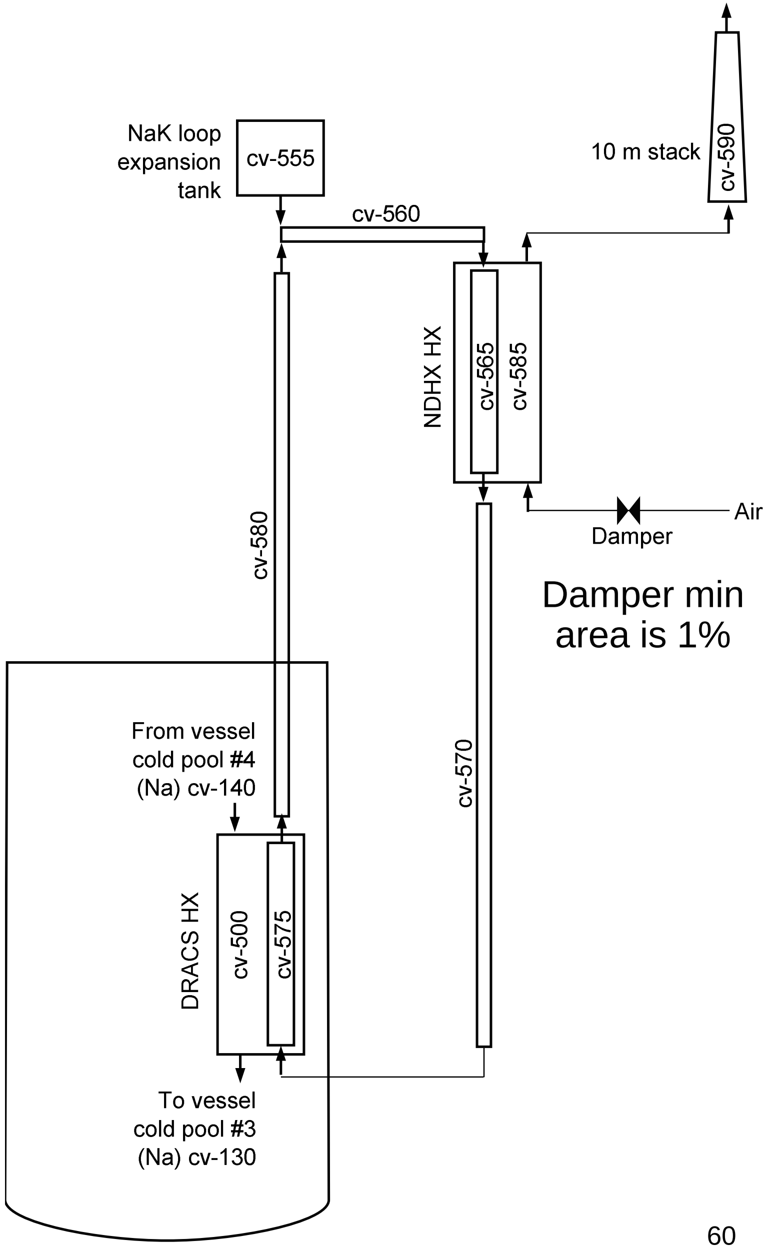
4 trains – 625 kW/train

- 0.25% of rated power per train (passive mode)
- Passive or forced circulation operation (only passive mode modeled)

Each train has 3 loops in series

- Cold pool primary coolant circulates through DRACS heat exchanger
- A Na-K secondary side loop transfers heat from the DRACS HX to the natural draft heat exchanger (NDHX)
  - Pump-driven or passive (only passive flow modeled)
- Air flows through the NDHX to the plant stack
  - Fan-driven or passive (only passive flow modeled)

Start-up: Damper on air flow springs open



Containment

Containment dome

- Defense in depth feature – radiological release and external challenges

Nitrogen-inerted guard vessel surround the reactor vessel

- Contains sodium leak and maintains sodium level above the fuel

Reactor cavity and air gap (i.e., not a safety system)

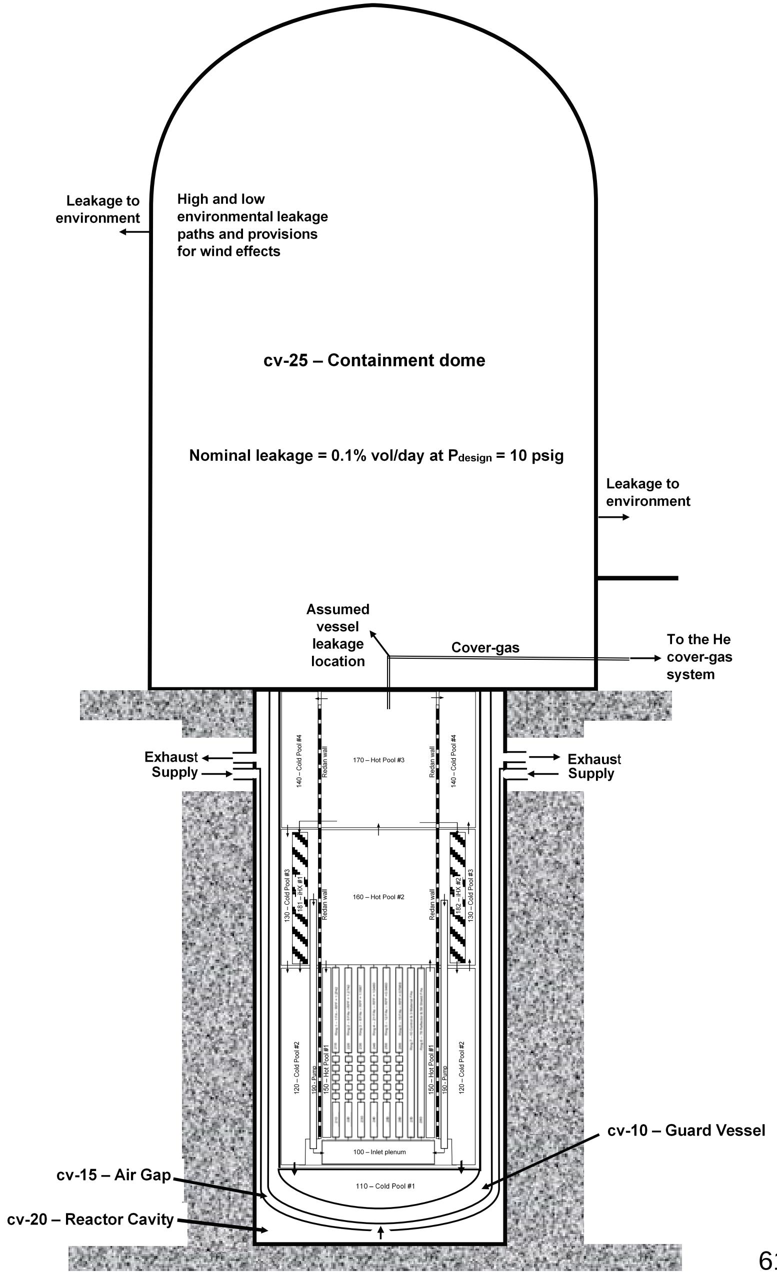
- Forced air cooling of concrete

Argon cover-gas above the reactor hot and cold pool regions

- System piping is not specified in the design description
- Assumed to be the source of radionuclide leakage

Leak rate is consistent with LWR containments

- 0.1% vol/day at 10 psig (design pressure)
- Dome = 5,580 m^3



MELCOR model inputs

Equilibrium inventory and decay heat from SCALE

Radial and axial power profiles from SCALE

Reactivity feedbacks from ANL ABTR report [ANL-AFCI-173]

U-10Zr fuel properties from INL [INL/JOU-17-44020]

HT-9 cladding and duct properties from [Leibowitz] & Bison [Hales]

Scenarios

Unprotected transient over-power (UTOP)

- Withdraw of highest worth control rod
- Failure of the control rods to insert

Unprotected loss-of-flow (ULOF)

- Trip of primary and intermediate sodium pumps
- Failure of the control rods to insert

Single blocked assembly

- Single assembly blocked
- Leak from the cover gas piping into the containment

Unprotected transient over-power (UTOP)

Initial and boundary conditions

- Highest worth control rod (0.9$) withdraws over 51 sec at mechanically-limited rate
- Reactor safety control rods fail to insert
- Primary and intermediate pumps continue to operate
- Intermediate heat exchanger remains operating

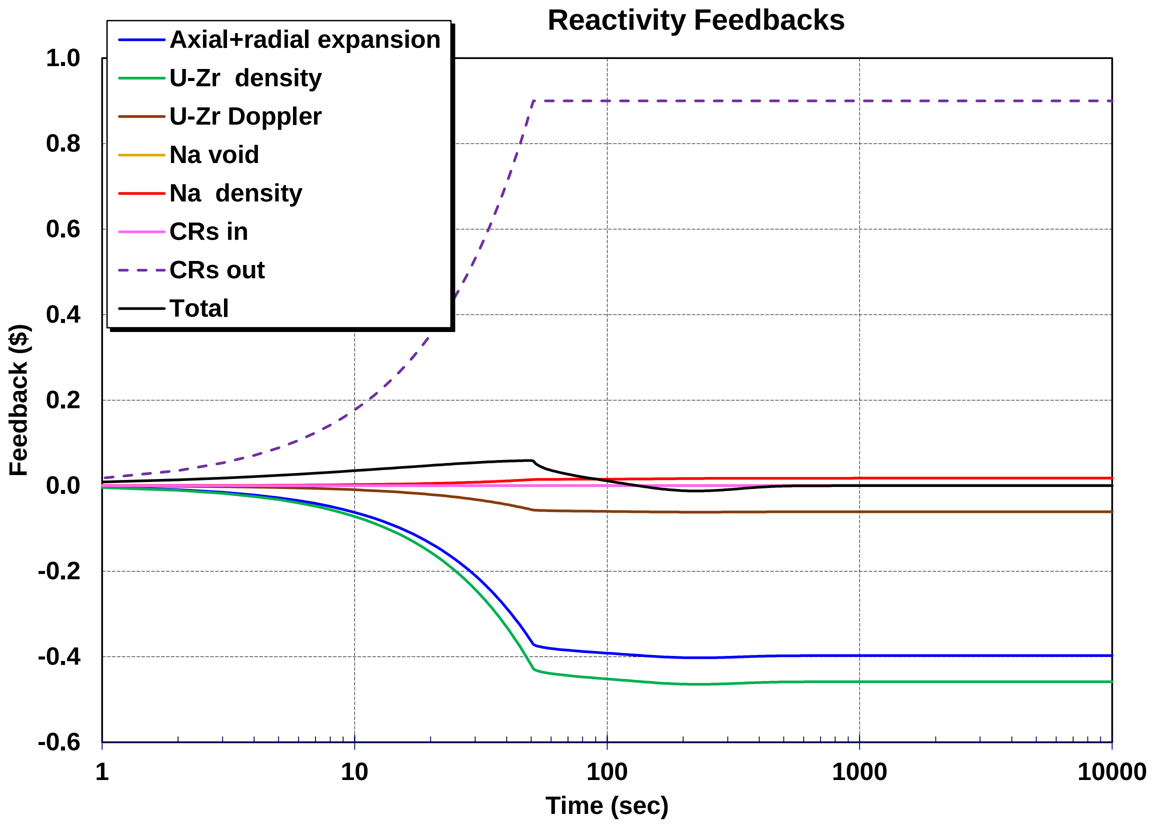
Sensitivity analysis on additional reactivity addition

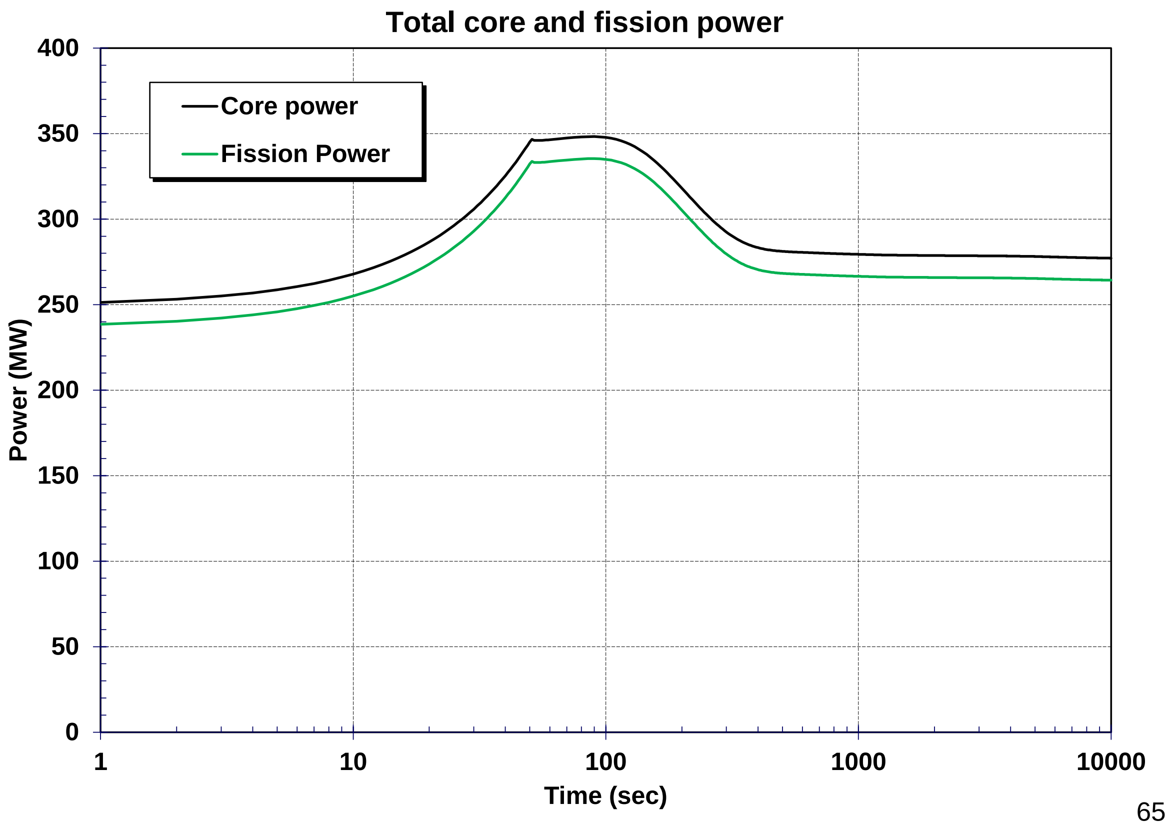
- Additional sensitivity calculations at 1.5$, 2.0$, & 2.5$, and 3.0$
- Sensitivity calculations on intermediate loop heat removal (i.e., limited to ~280 MW or unlimited)

UTOP – Withdraw of highest-worth CR

- The highest-worth CR withdraws over 51 sec to insert 0.9$.
- The net reactivity initially increases but is subsequently balanced by the negative feedbacks after the CR is withdrawn

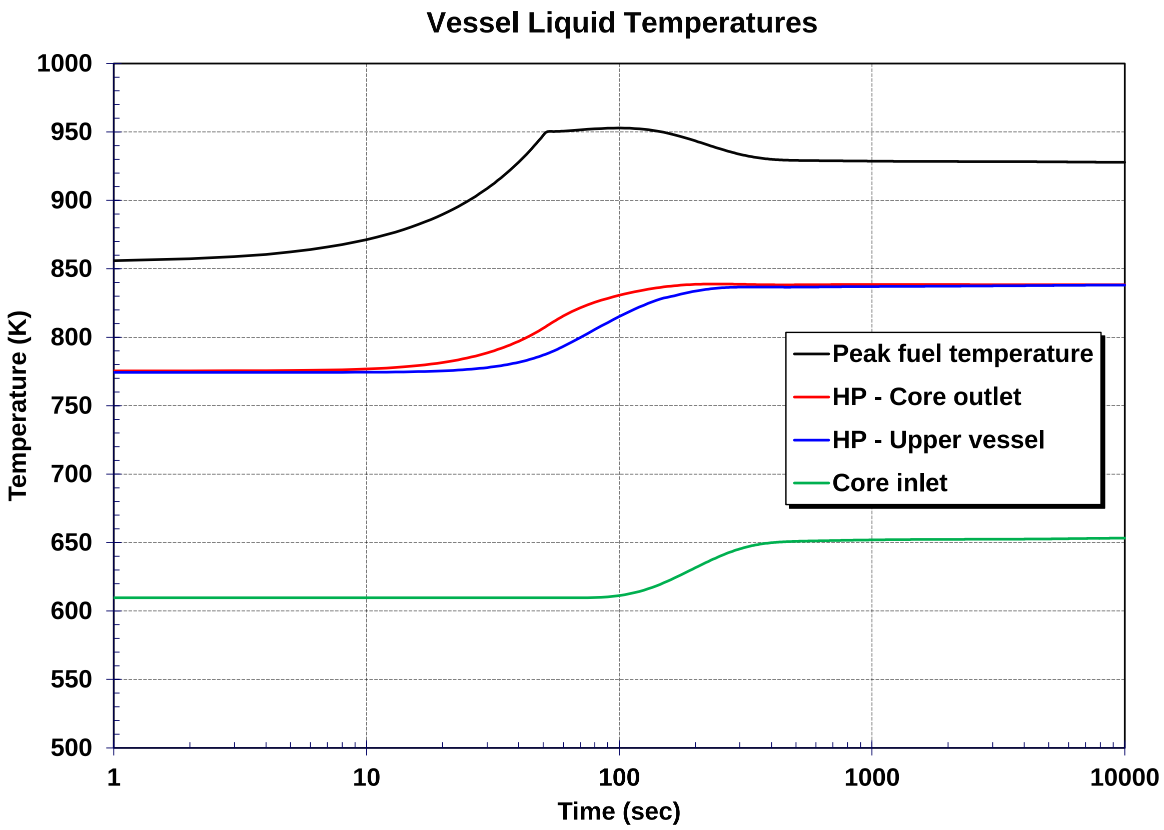
- The core power rises to 346 MW in response to the reactivity insertion but subsequently drops in response due to the strong negative fuel feedback.
- The long-term power stabilizes at 280 MW
  - The maximum intermediate loop heat removal was assumed to be limited to (280 MW) ~112% of rated





UTOP – Withdraw of highest-worth CR

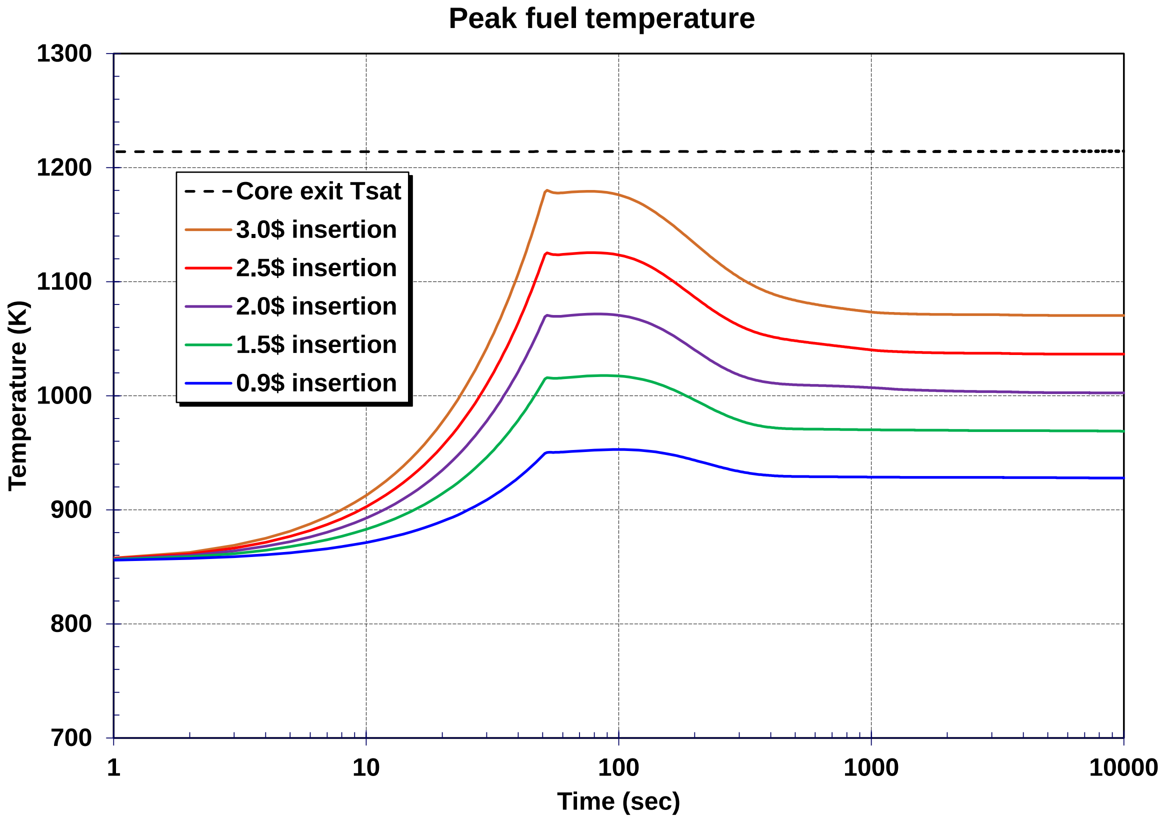
- A 952 K peak fuel temperature occurs at 100 sec due to the CR withdraw and reactivity insertion
  - The reactivity feedback and the fuel temperature adjust to match the secondary heat removal
- The hot pool at the core exit has a ~64 K temperature rise, which increases the core inlet temperature
- Large margin to U-10Zr fuel melting (1623 K)

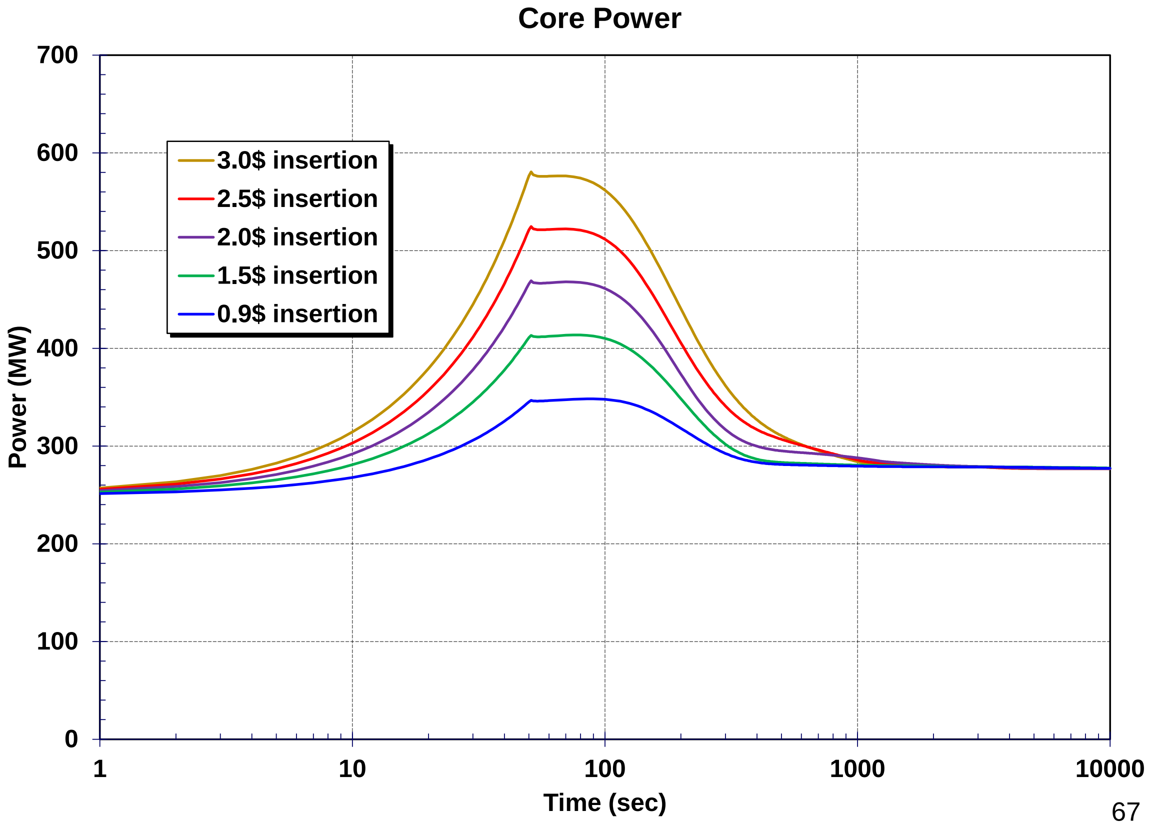


UTOP – CR worth sensitivity

- A larger reactivity insertion leads to successively higher peak fuel temperatures
- The peak fuel temperature response is approaching the sodium saturation temperature (~1215 K) in the 3.0$ case

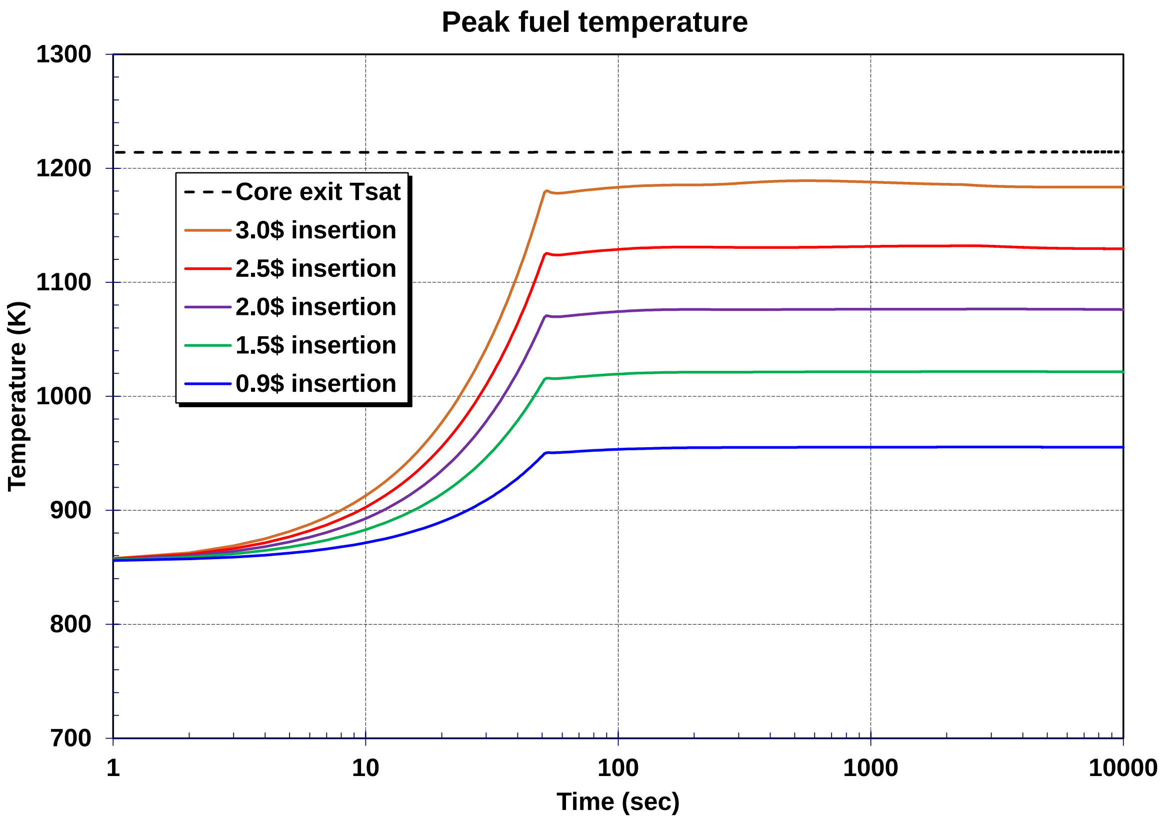
- A larger reactivity insertion leads to corresponding higher peak core powers
- The long-term core power reflects the assumed capacity of the intermediate loop heat removal (~280 MW)
- The core inlet temperature increases with higher reactivity insertions

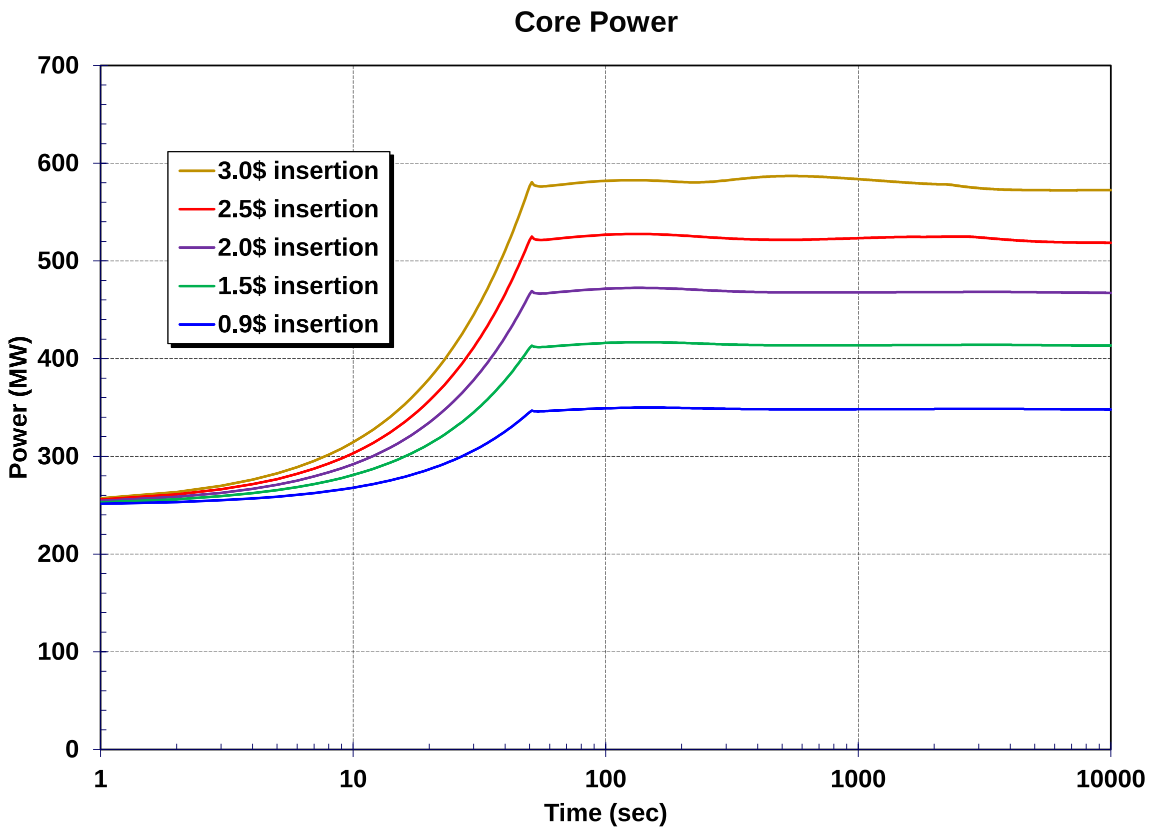




UTOP – Unlimited intermediate loop heat removal

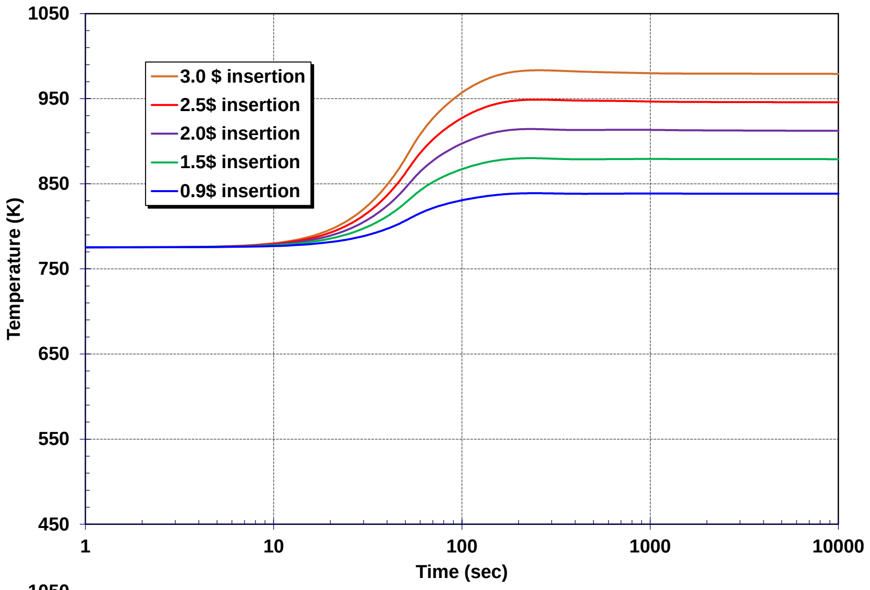
- The fuel temperature does not decrease following the reactivity addition since the control rods remain withdrawn
  - The core inlet temperature remains approximately constant in all cases
- The core inlet temperature remains near the rated condition but the exit temperature and the corresponding core temperature rise settles to offset the insertion of additional reactivity
  - Higher core power → higher fuel temperature → higher intermediate loop heat removal requirements

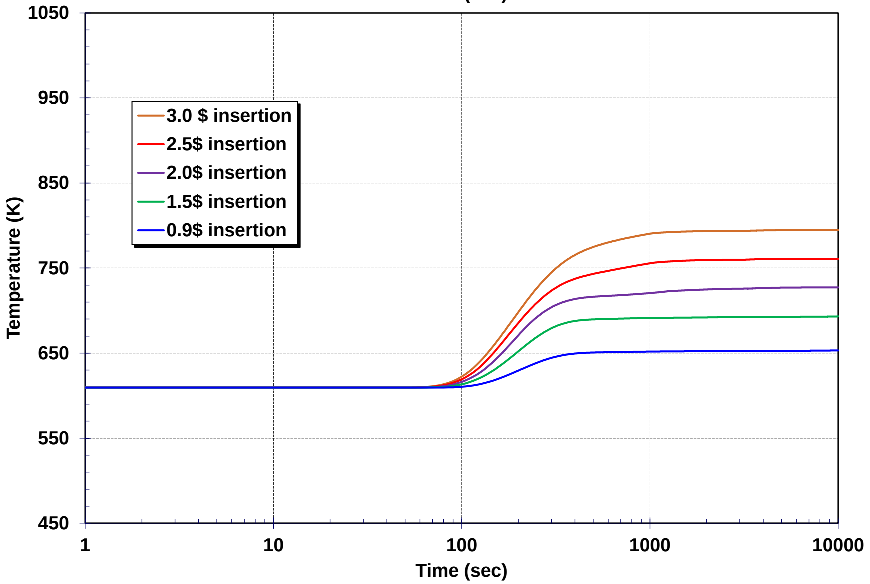




UTOP – Unlimited intermediate loop heat removal

Limited Heat Removal Capacity (~112%)

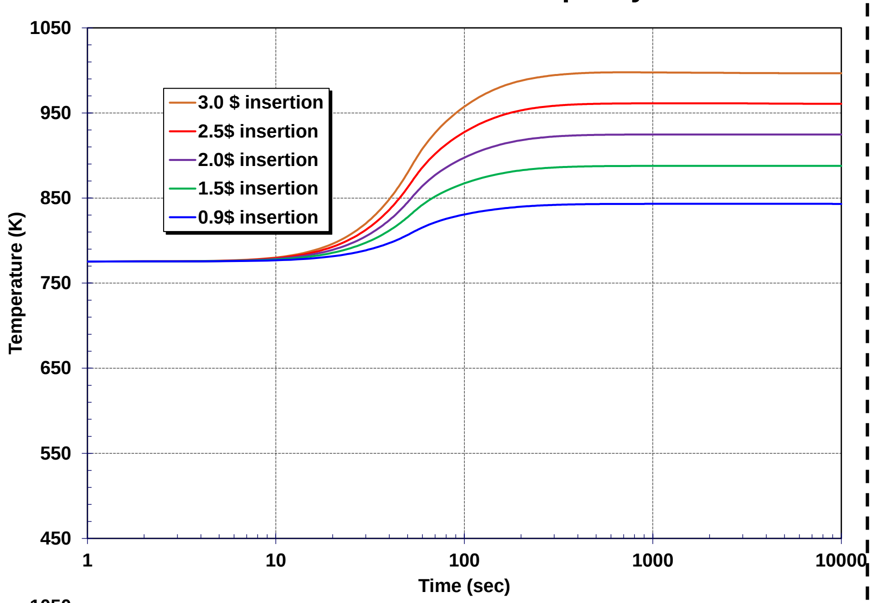


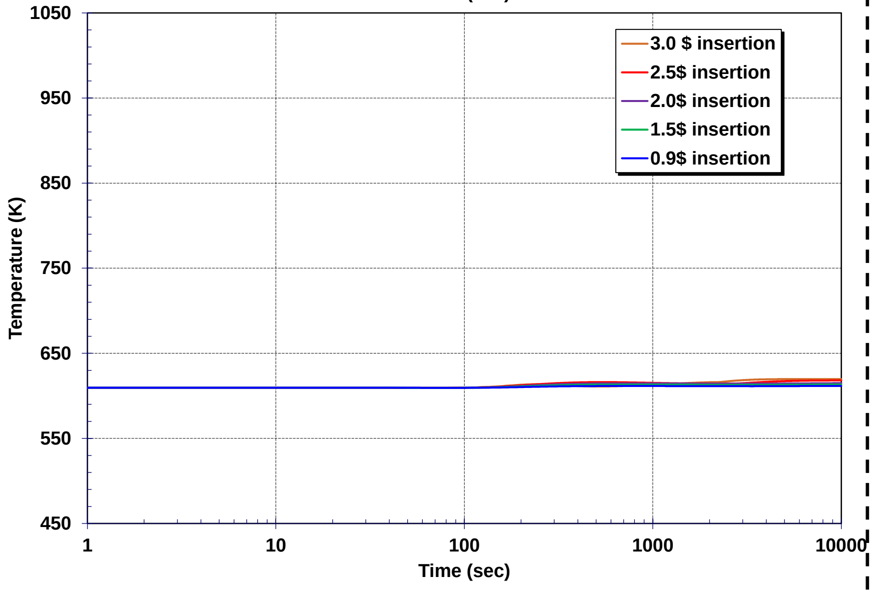


Core Outlet Temperature

Core Inlet Temperature

Unlimited Heat Removal Capacity





Unprotected loss-of-flow (ULOF)

Initial and boundary conditions

- Primary and intermediate pumps trip resulting in no secondary heat removal
- Reactor safety control rods fail to insert
- 4 DRACS trains are available in passive mode

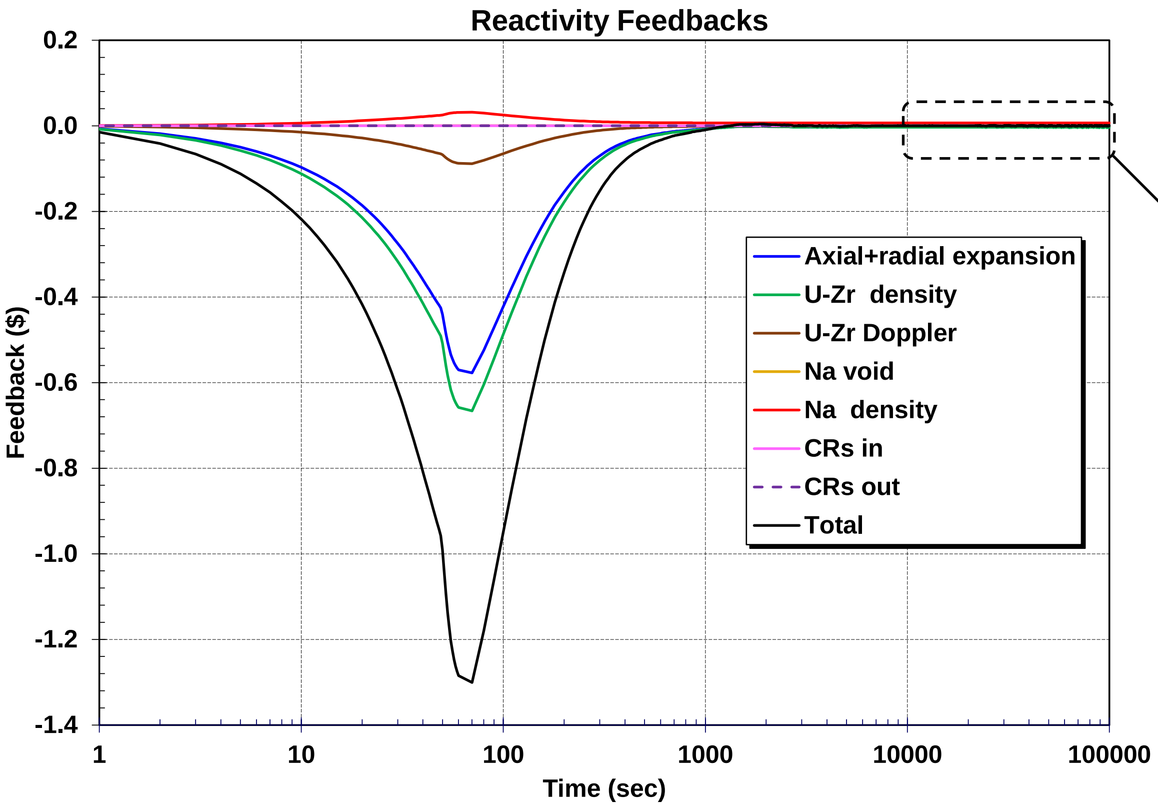
Sensitivity analysis on DRACS availability

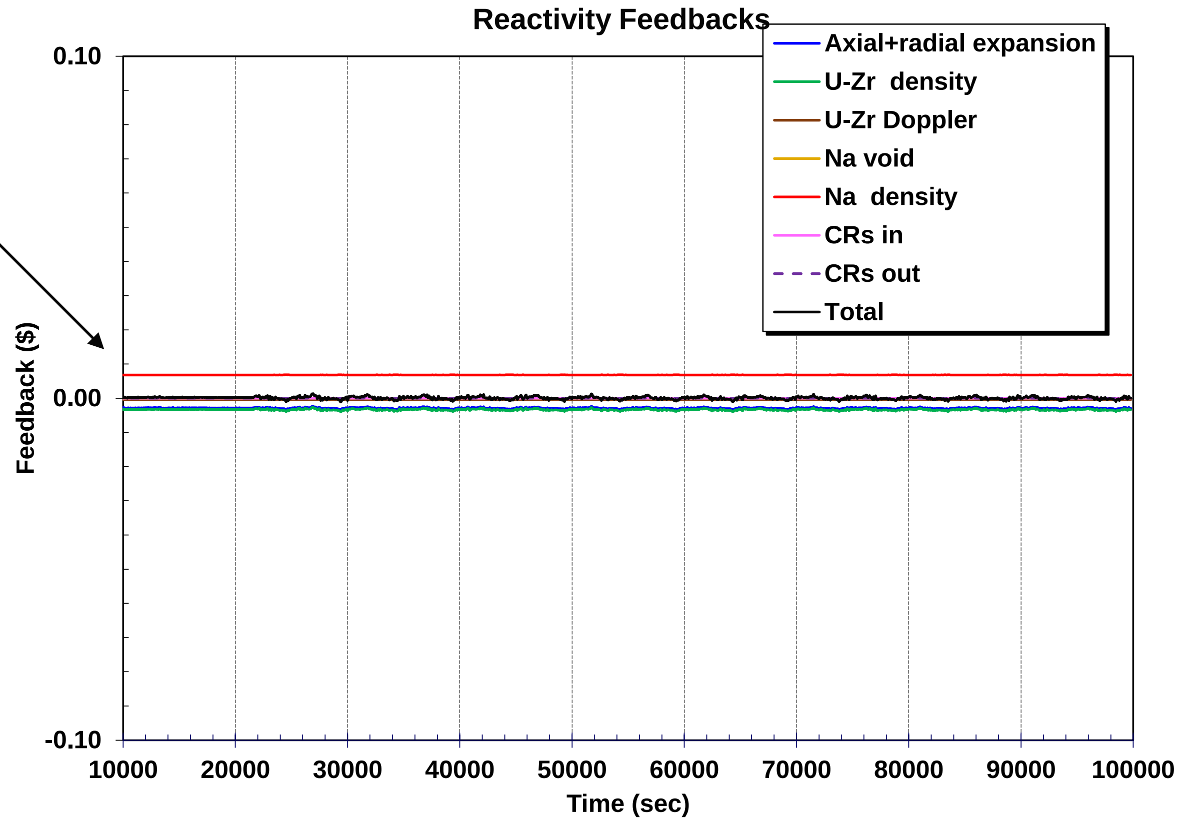
- 0, 1, 2, and 3 DRACS trains available

ULOF

The initial fuel heatup has strong negative expansion, fuel density, and fuel Doppler fuel feedbacks that greatly offsets the positive sodium density feedback that shuts down fission

The net reactivity oscillates near zero after 1000 sec





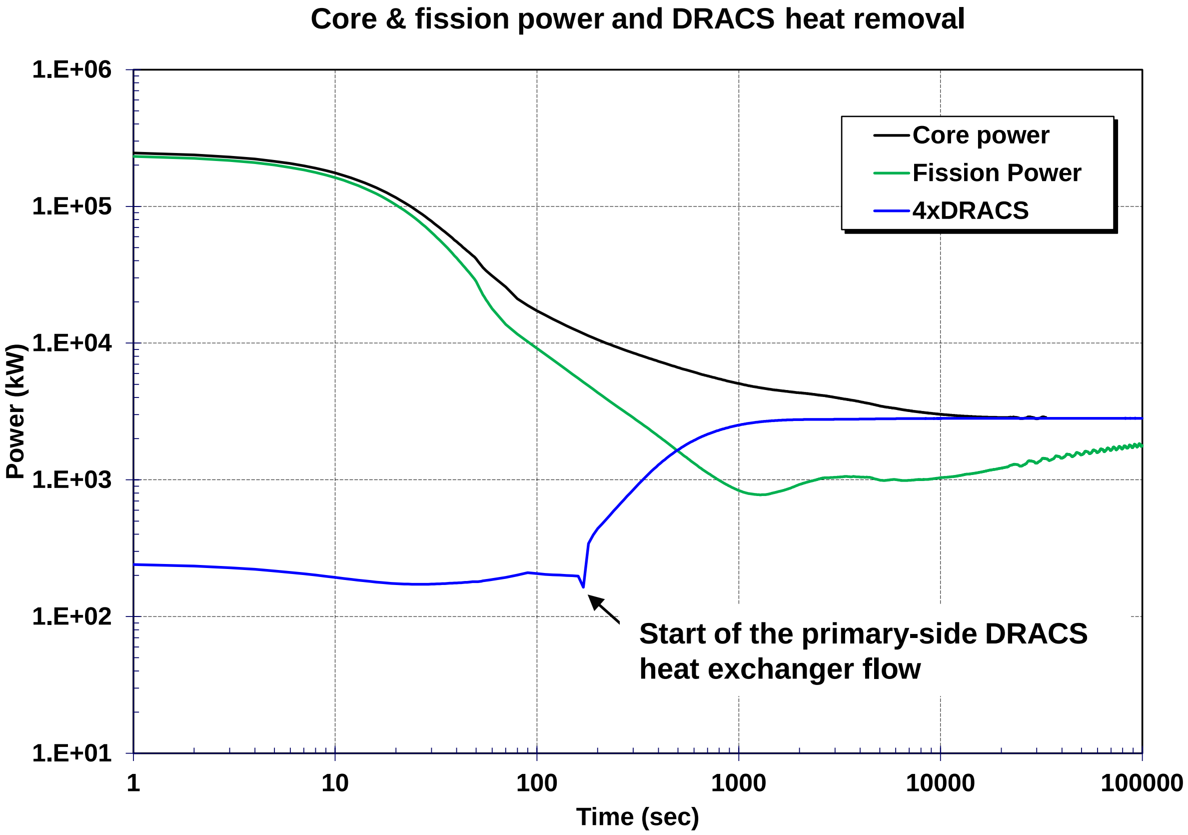
ULOF

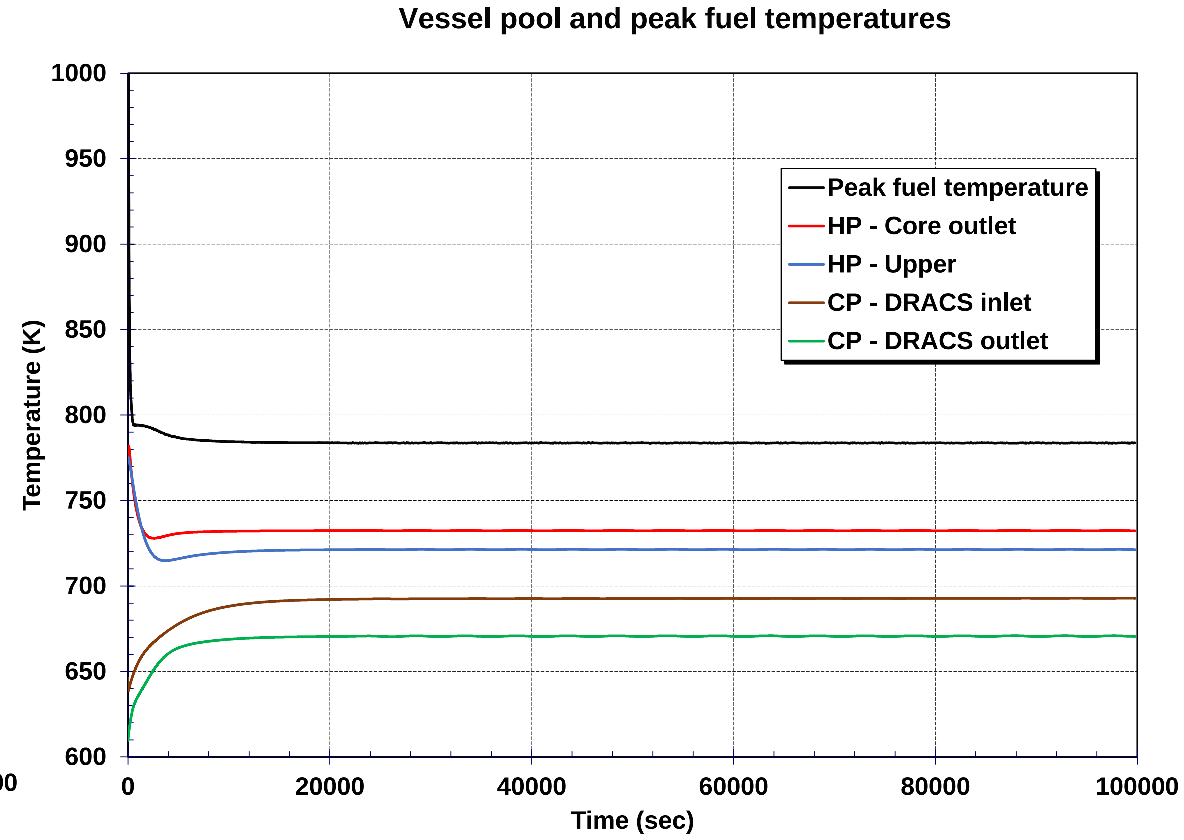
The long-term core power matches the DRACS heat removal rate after 20,000 sec (5.6 hr)

The fission power is 1000 kW at 10,000 sec and gradually increases to offset the decrease in decay heat

The fuel and vessel liquid sodium temperatures quickly stabilize

The natural circulation flow moves heat from the core, through the iHXs to the cold pool, and through the DRACS





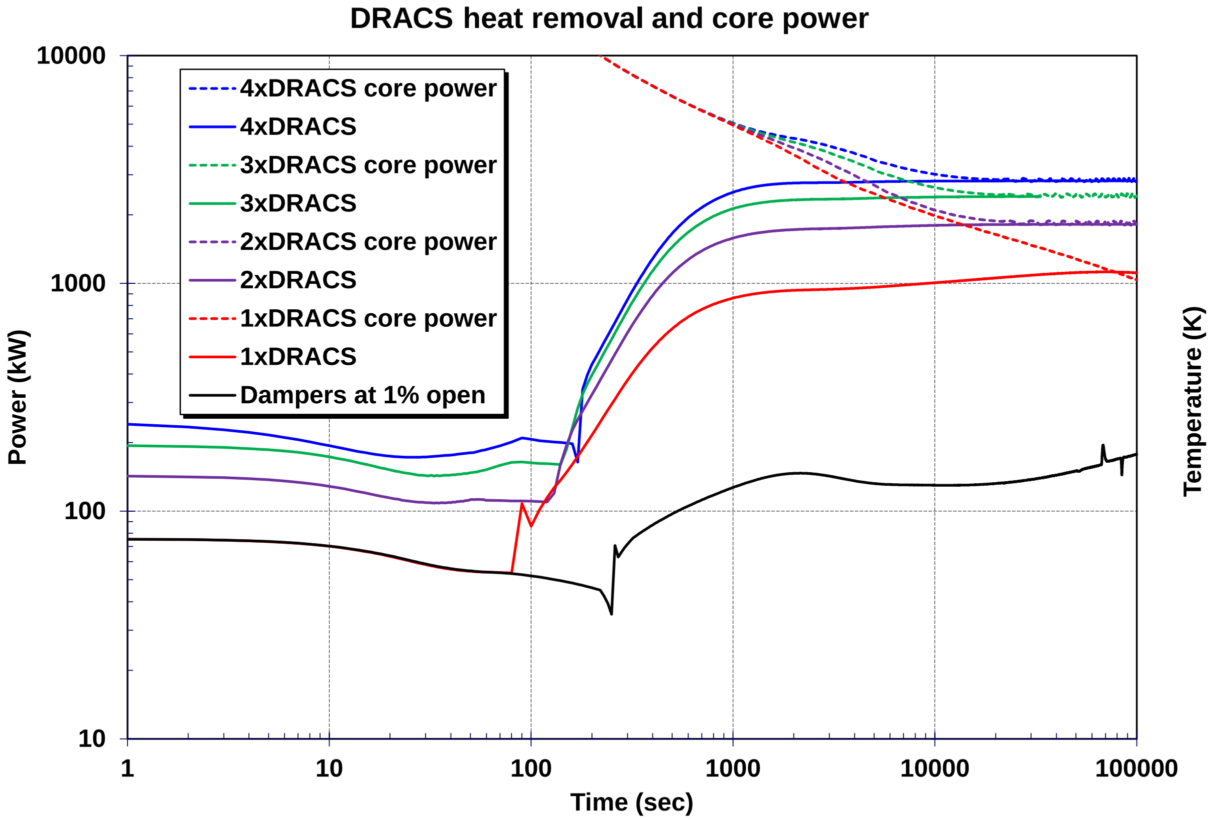
ULOF – with variable DRACS sensitivity

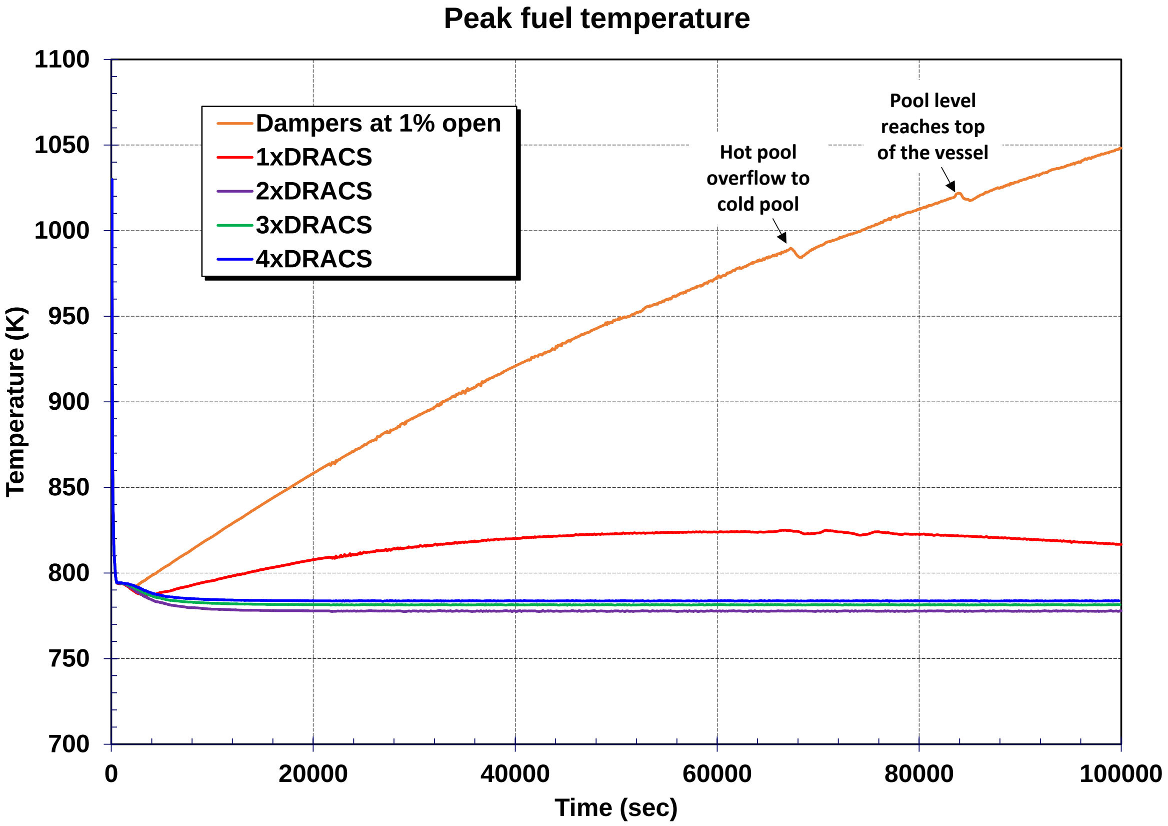
- Core power eventually converges on the DRACS heat removal rate
- Dampers are normally 1% open

1xDRACS case shows a small heatup but other DRACS cases have similar responses

- Thermal inertia of the DRACS and vessel mitigate heatups

Expansion of sodium leads to hot to cold pool spill-over and eventually a filled vessel in 1% damper case

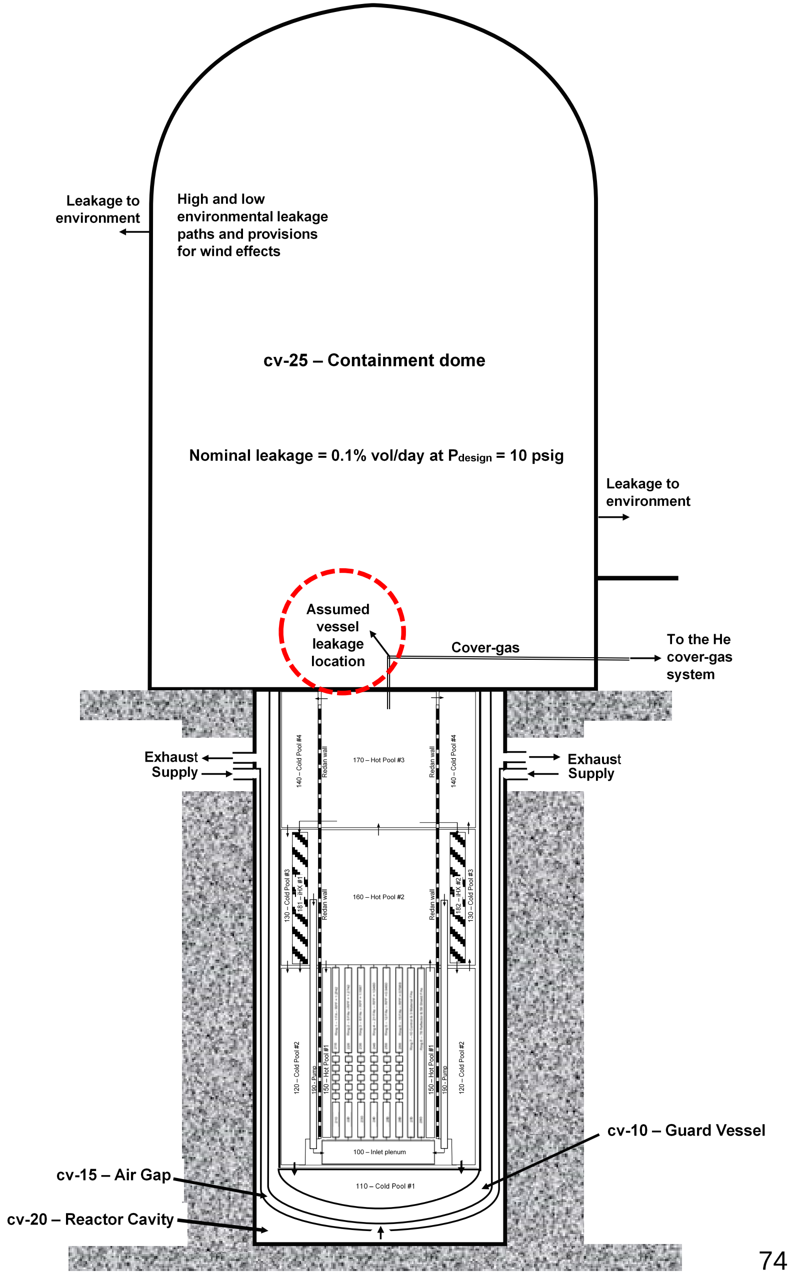




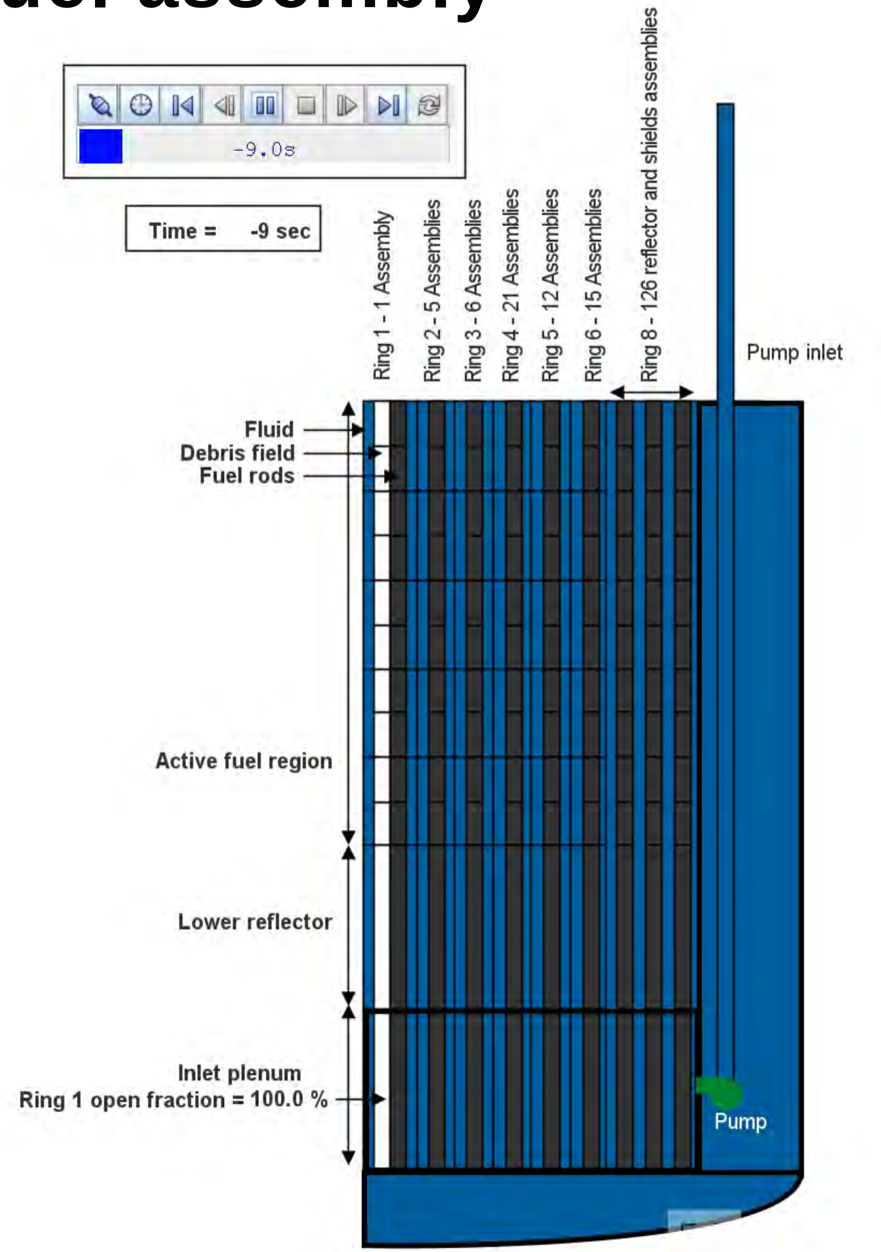
Single blocked fuel assembly

Initial and boundary conditions

- Inlet to a fuel assembly is blocked
- Primary and intermediate pumps remain running
- Control rods are assumed to insert after an off-gas high-radiation signal
- The cover gas system leaks in the containment
  - Assumed radionuclide release pathway



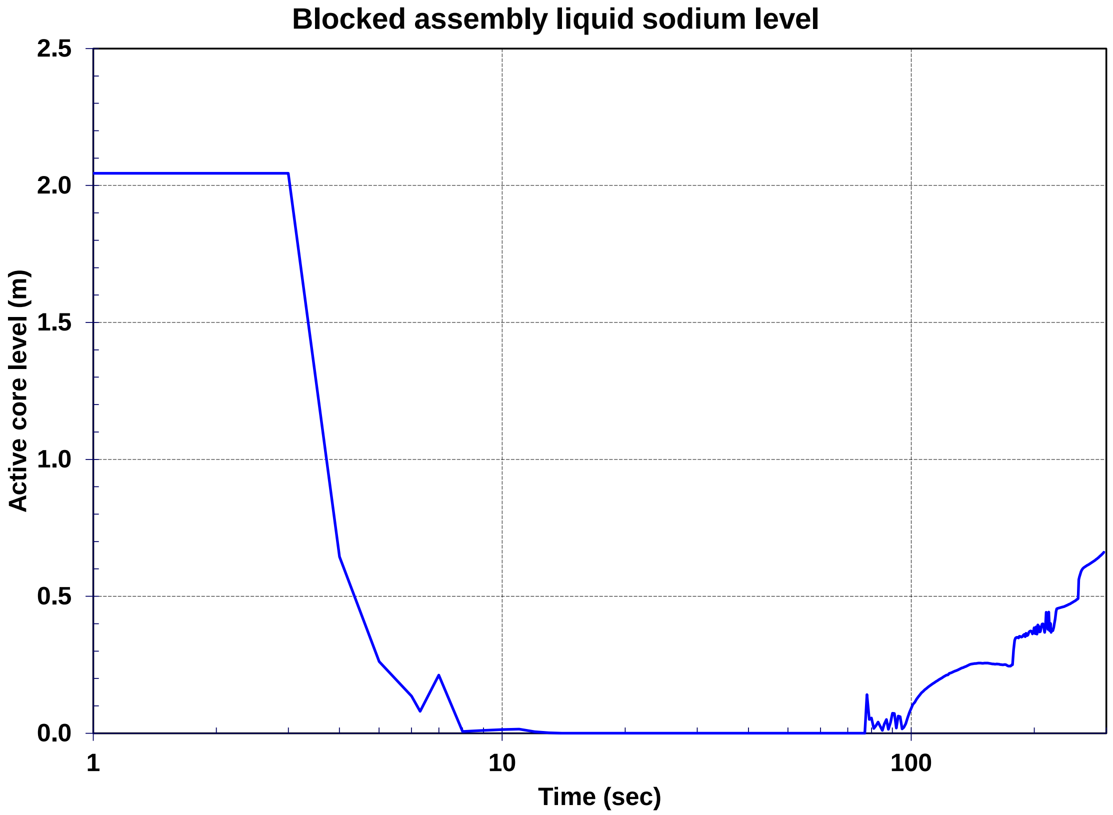
Single blocked fuel assembly

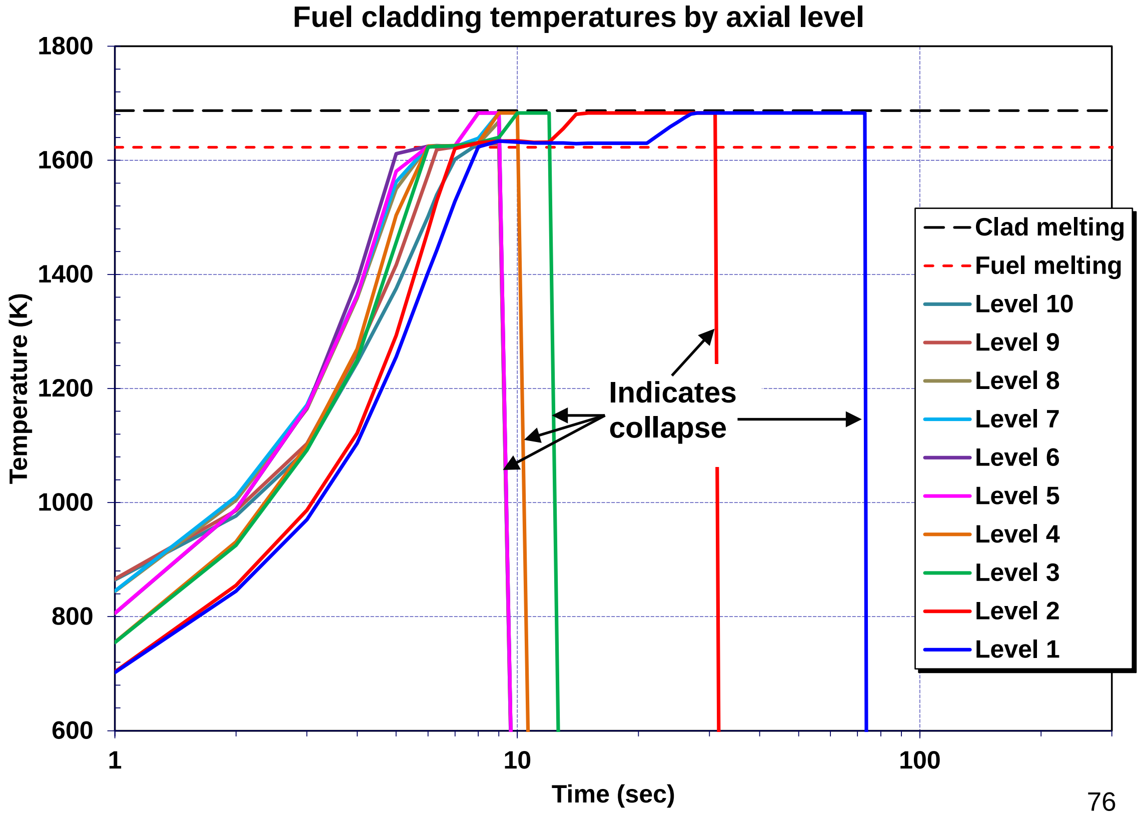


Single blocked fuel assembly

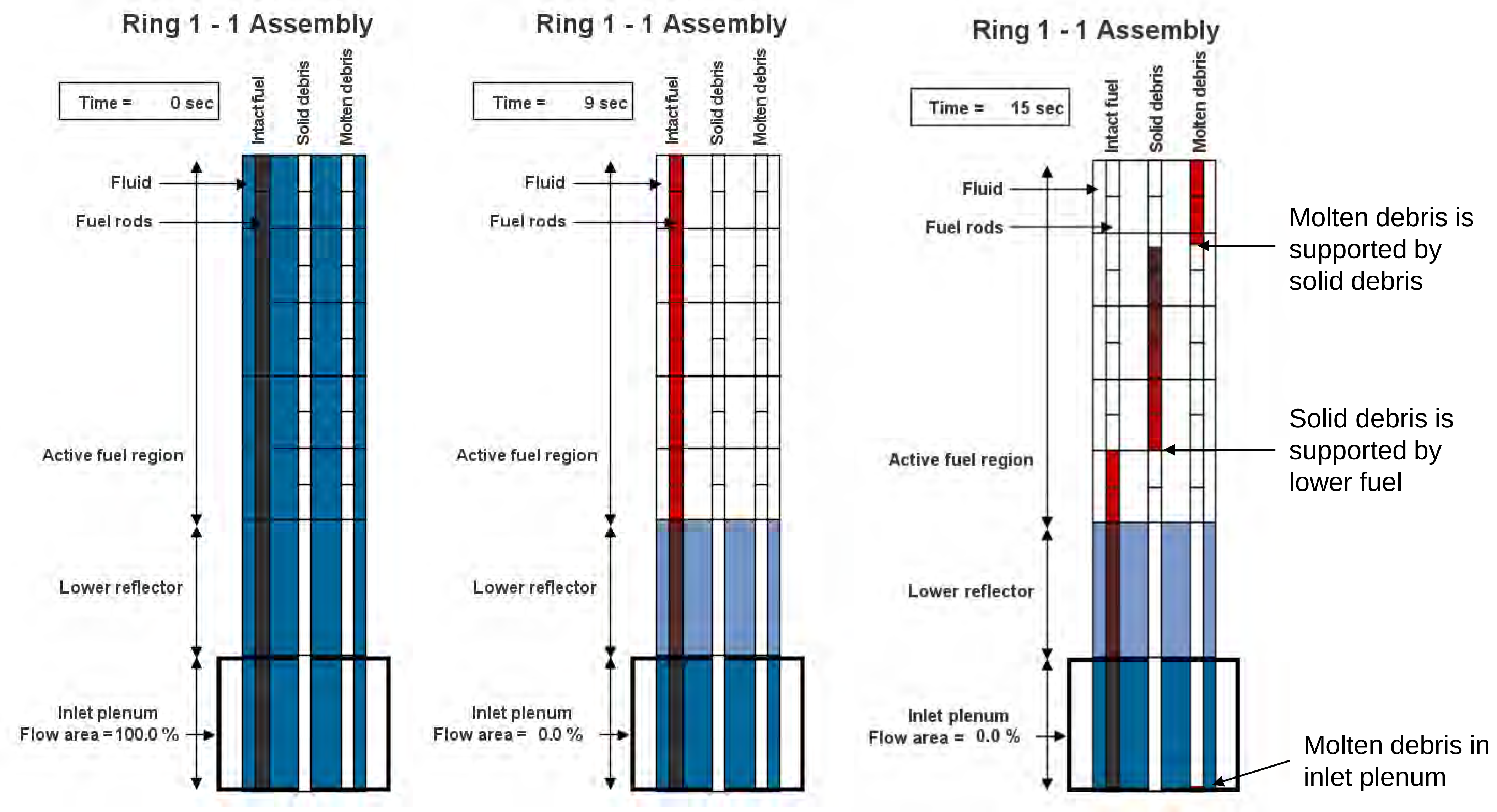
- The fluid in the duct starts voiding within 3 seconds
- The assembly sodium is boiled and expelled within ~10 sec

- The fuel cladding temperature responses (below) also indicate the fuel temperature response
- The cladding temperature rise pauses while the fuel melts and then increases to the cladding melting temperature
- The cladding melts and collapses when the minimum thickness reaches a structural integrity limit



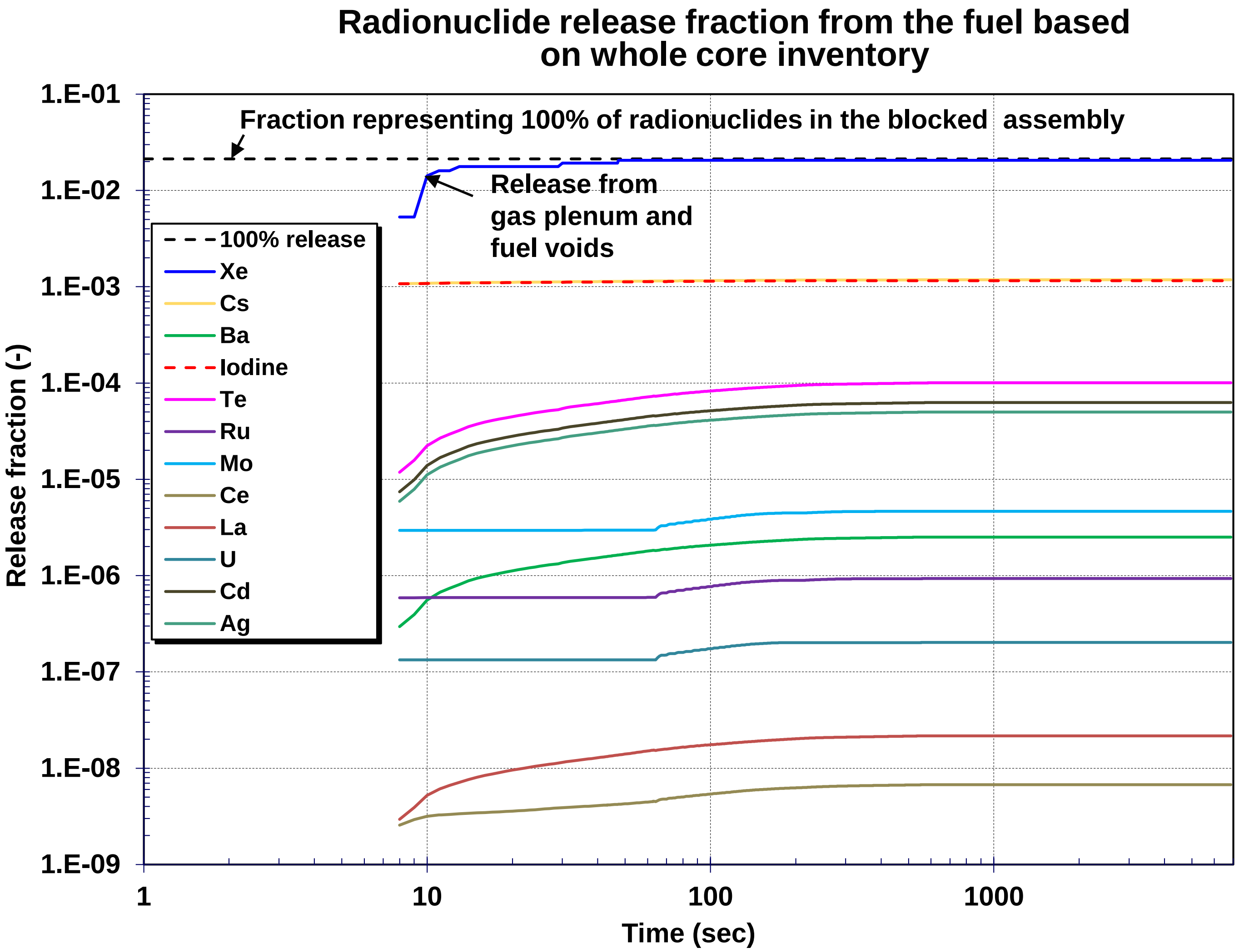


Single blocked fuel assembly



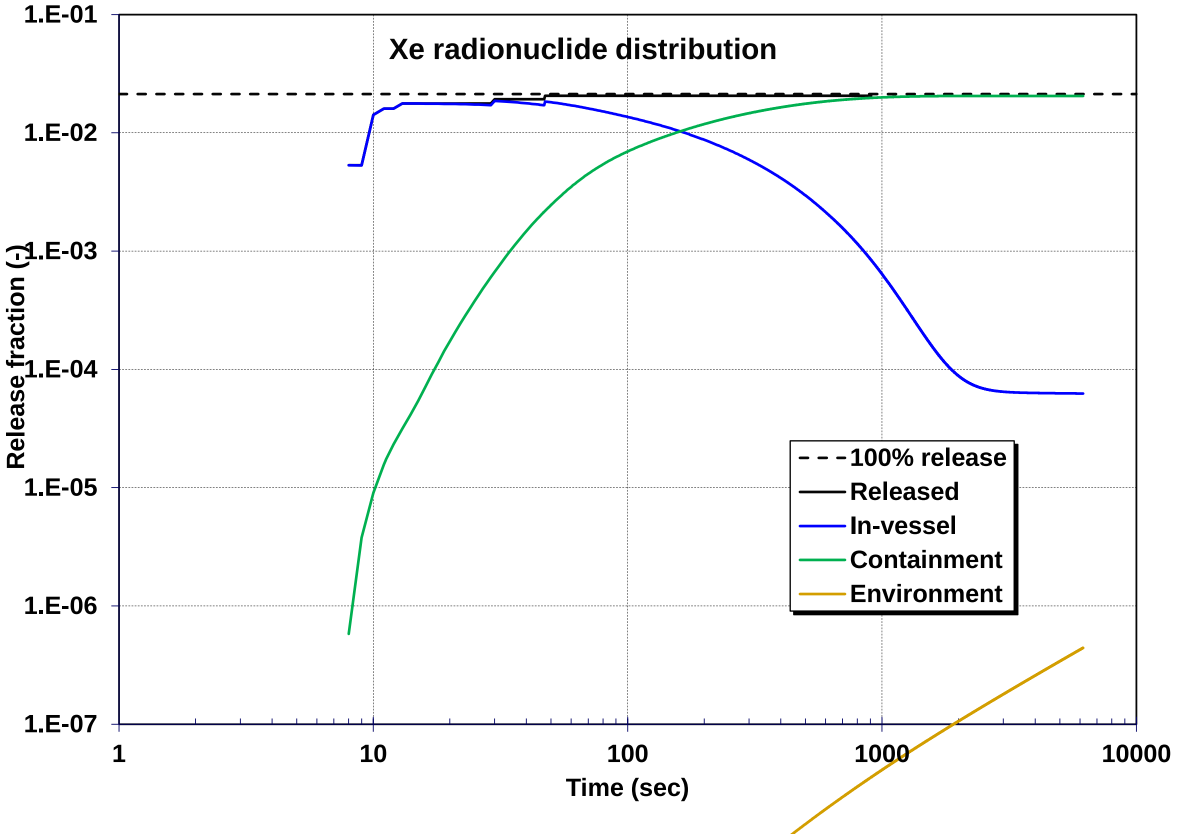
Single blocked fuel assembly

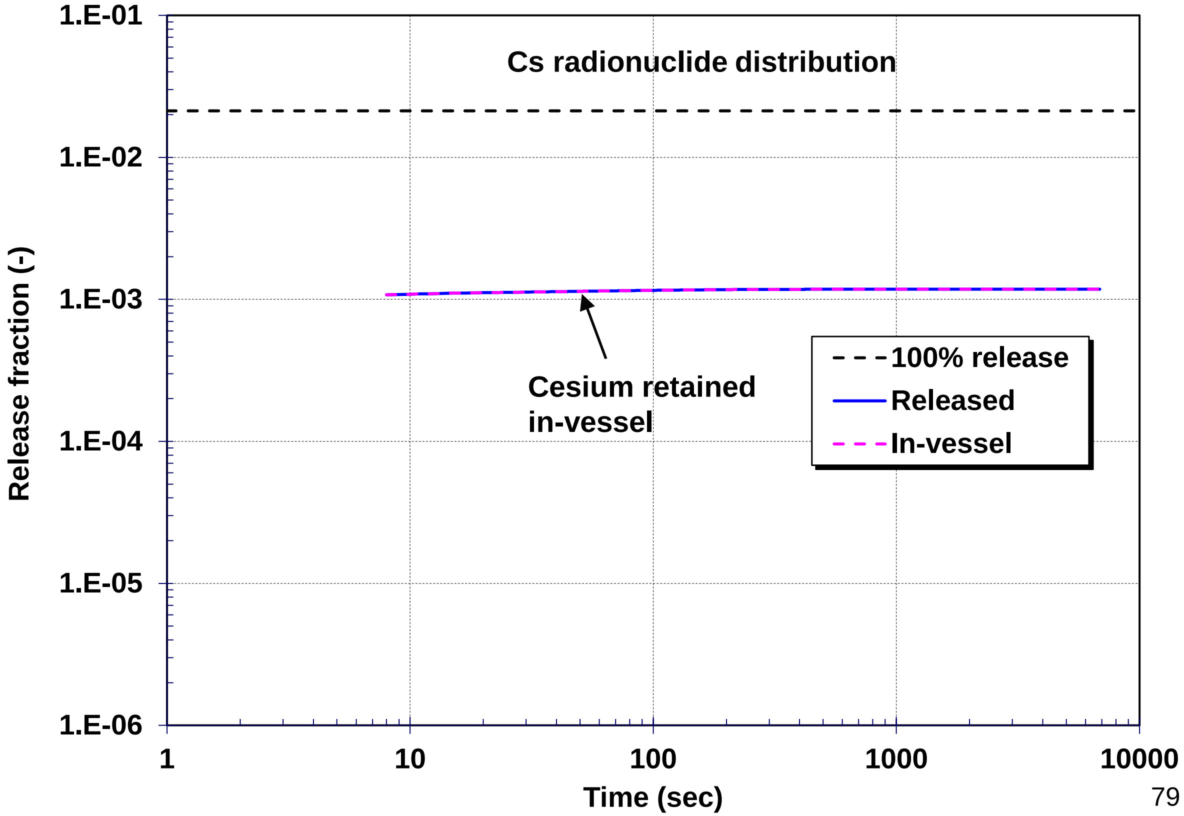
- After the cladding failure, there is a prompt release of the plenum gas inventory followed by thermal releases from the hot debris
- The analysis assumed blockage of a high-powered center assembly with approximately 2.2% of the core radionuclides
  - 97% of the noble gases
  - ~6% of iodine and cesium



Single blocked fuel assembly

- Xe bubbles through the hot sodium pool above the core to the gas space.
- Leakage rate through the failed off-gas line to the containment
  - Assumed the sweep flow of 1 reactor gas space change per hour persisted during the transient
  - Xe environmental release is very small due to the large containment volume and the low leak rate
- The cesium and other radionuclides retained in the sodium





MELCOR Summary

MELCOR SFR Summary

- MELCOR capabilities were demonstrated
  - New phenomenological modeling added to MELCOR for SFRs
- Capabilities for a broad range of SFR accident scenarios (e.g., UTOP, ULOF)
- Key physics considered
  - Neutronics
  - Liquid metal thermal hydraulics
  - Core heat-up and degradation
  - Fission product release
- Future work
  - Modeling improvements and enhancements
  - Fuel cycle analysis (Volume 5)

Concluding remarks