Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

NRC's simulation capabilities supporting criticality, reactor physics, decay heat, and shielding for metallic fueled non-LWRs

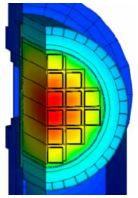
Andy Bielen, Ph.D.

Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

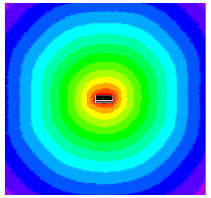
Objectives

- NRC's simulation capabilities supporting nuclear fuel safety for metallic fuel designs
  - Decay Heat
  - Neutron Multiplication & Criticality
  - Shielding and Radiation Protection
- Overview of data availability, gaps, and where additional data would be beneficial

Nuclear Physics Considerations for TRISO/SFR Spent Fuel Safety



Decay Heat



Shielding and Radiation Protection

Neutron Multiplication and Criticality

NRC Regulations limit radiation dose under all phases of the fuel cycle:

- Direct radiation dose
- Radioactive material releases
- Inadvertent criticality

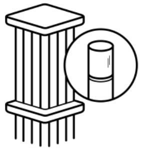
Computer codes used to determine:

- Irradiated fuel composition for nuclides that contribute to:
  - Direct radiation dose and dose from radioactive material releases
  - Decay heat
  - Determination of criticality safety (k_{eff})
- Radiation dose and k_{eff}

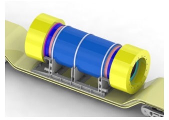
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

10 CFR 50/52 – Power Plants



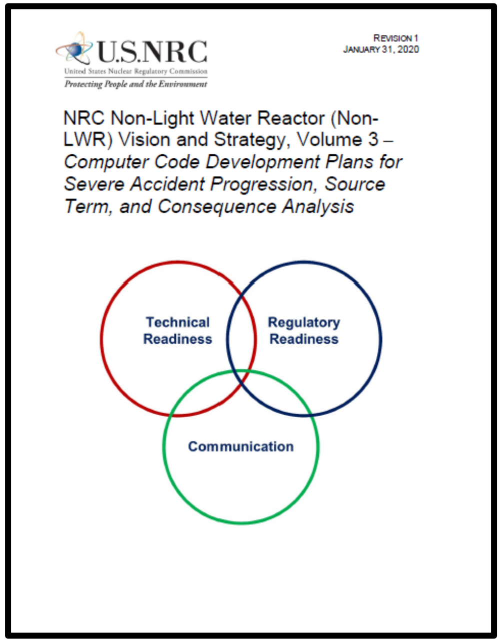
10 CFR 70 – Fuel Cycle Facilities



10 CFR 71 – Transportation

10 CFR 72 – SNF Storage

Non-LWR Source Term & Fuel Cycle Demonstration Projects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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*

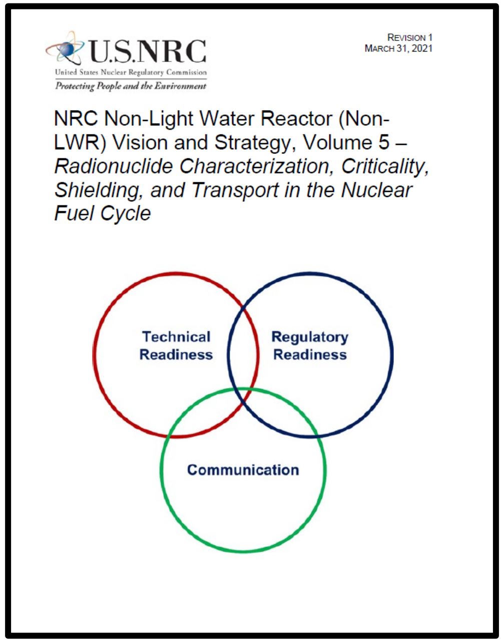
[Source Term]

NRC's comprehensive neutronics package

- Cross-section processing
- Decay heat analyses
- Criticality safety
- Radiation shielding
- Radionuclide inventory & depletion generation
- Reactor core physics

NRC's comprehensive severe accident progression and source term code

- Accident progression
- Thermal-hydraulic response
- Core heat-up, degradation, and relocation
- Fission product release and transport behavior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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*

[Fuel Cycle]

Non-LWR demonstration projects improve and validate SCALE & MELCOR for simulating non-LWRs for severe accident progression and fuel cycle analyses.

Using SCALE to Calculate Non-LWR Neutronics Quantities of Interest

Decay Heat

- SCALE/TRITON is used to generate specific ORIGEN reactor libraries; functionally bounds fuel enrichment and burnup.
- SCALE/ORIGAMI is used to obtain the spent fuel inventories; uses ORIGEN to compute detailed irradiated and decayed isotopic compositions.

Criticality Safety

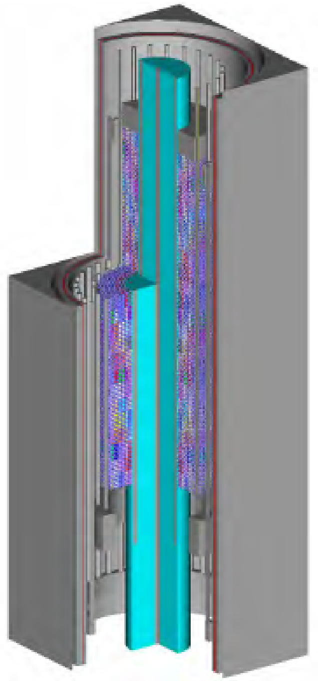
- SCALE/CSAS is used to perform criticality safety analyses. CSAS is a sequence that uses Monte Carlo transport codes KENO or Shift.
- Used to determine the multiplication factor of any system.

Shielding & Dose

- SCALE/MAVRIC is used to perform the shielding and dose analyses.
- Uses the radiation source term & radionuclide inventories generated from SCALE/TRITON or SCALE/ORIGAMI.

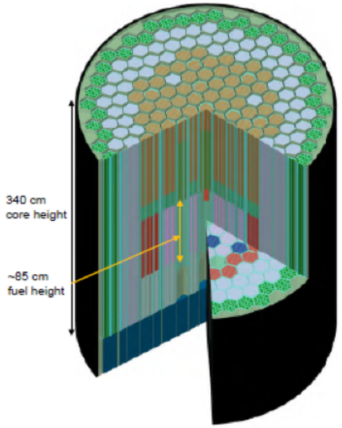
Non-LWR Reference Models

High-Temp. Gas Cooled Reactor	Sodium-Cooled Fast Reactor	Molten Salt-Cooled Reactor	Molten Salt-Fueled Reactor	Heat Pipe Reactor



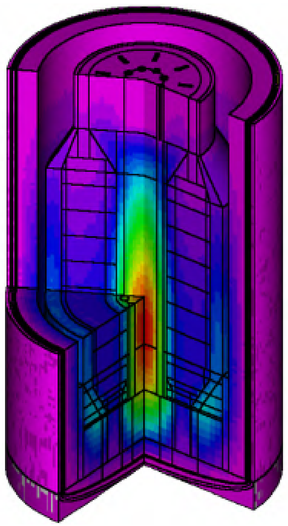
PBMR-400

- 400 MWth reactor, graphite moderated
- Helium-cooled & TRISO-particle pebble-fueled at 10 wt.% U-235
- Fuel discharged at high burnup (90 GWd/MTIHM)



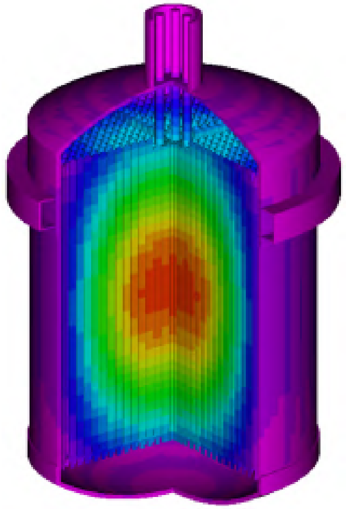
ABTR

- 250 MWth pool-type reactor, utilizing metallic U / HT-9 fuel rods
- Reactor fueled with U-Pu-Zr fuel slugs
- Liquid sodium coolant



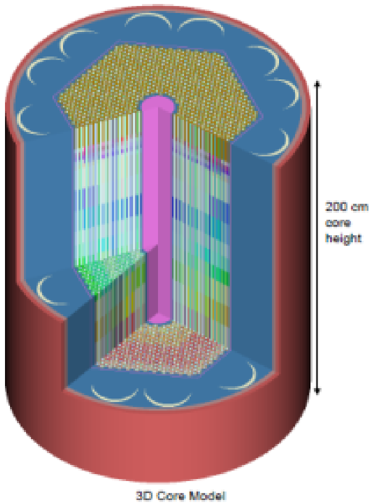
UCB Mk1 PB-FHR

- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling



MSRE

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with liquid dissolved fuel in molten salt (34.5 wt. % U-235)

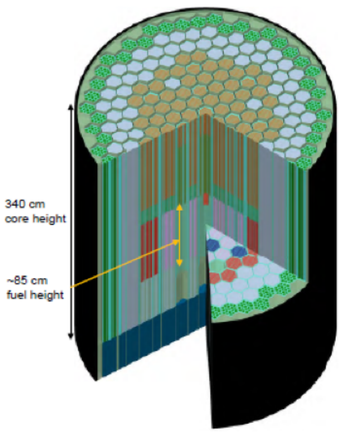


INL Design A

- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with UO_2 fuel (19.75 wt.% U-235)
- Reactivity controlled via control drums

Sodium Fast Reactor Workflows

Sodium-Cooled Fast Reactor



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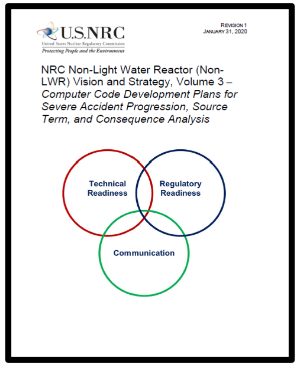
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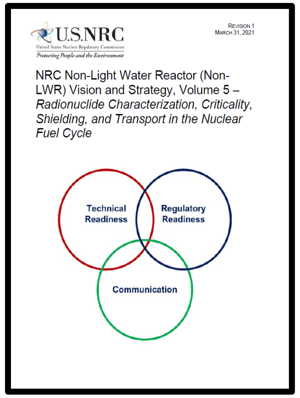
January 31, 2020

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*

Technical Readiness
Regulatory Readiness
Communication

Source Term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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*

Technical Readiness
Regulatory Readiness
Communication

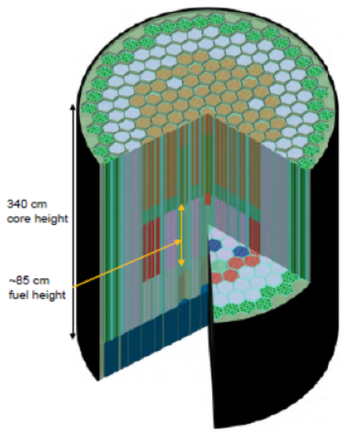
Fuel Cycle

Fuel Depletion, Decay Heat & Radionuclide Inventory Generation

- Decay Heat
- Criticality Safety
- Shielding & Dose

Fuel Depletion, Decay Heat, and Nuclide Inventory Generation

Sodium-Cooled Fast Reactor



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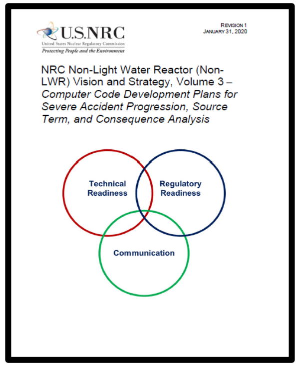
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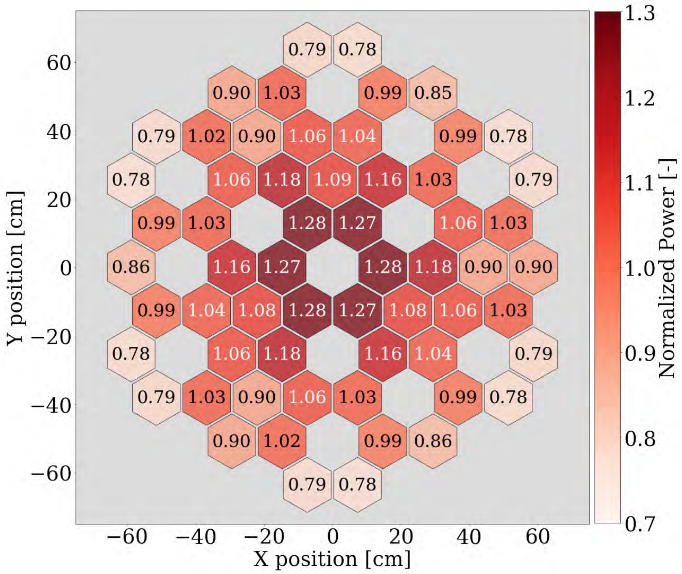
January 31, 2020
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*
Technical Readiness
Regulatory Readiness
Communication

Fuel Depletion, Decay Heat & Radionuclide Inventory Generation

Source Term

- SCALE/TRITON used for fuel depletion
  - Full core 3D continuous energy Monte Carlo physics
  - All fuel assemblies in the core depleted (Total of 60)
  - ORIGEN used to track >2,000 nuclides

- Radionuclide inventories used to support downstream analyses.
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses



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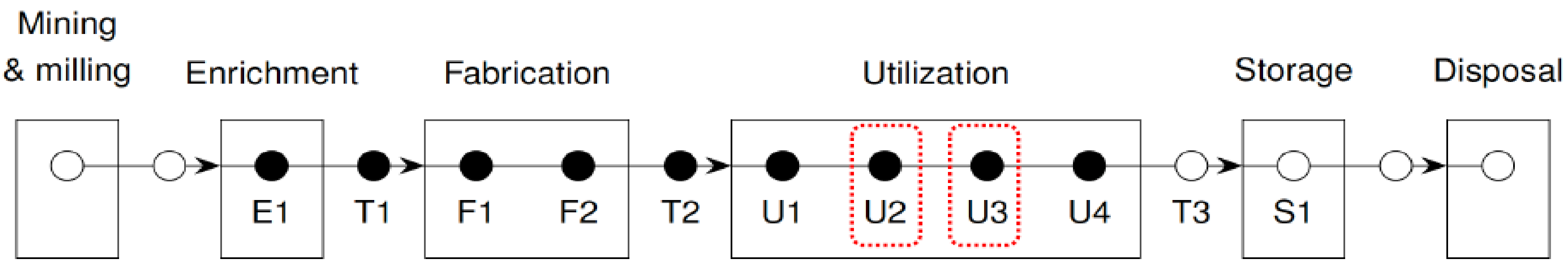
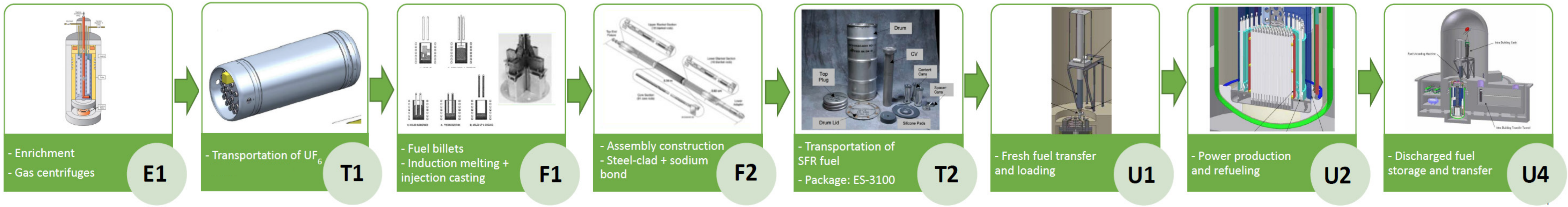
Sodium Fast Reactor Fuel Cycle

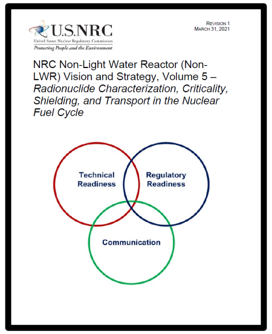


E1 – UF_6 enrichment
T1 – Transportation of UF_6 to fabrication facility
F1 – Fuel fabrication
F2 – Fuel assembly/pebble fabrication
T2 – Transportation of assemblies/pebbles/salt to plant
U1 – Fresh fuel staging/preparation/loading

U2 – Power production
U3 – Spent fuel pool/shuffle operations
U4 – On-site dry cask storage
T3 – Transportation of spent fuel to off-site storage
S1 – Off-site storage



Decay Heat, Criticality Safety, and Radiation Shielding / Dose

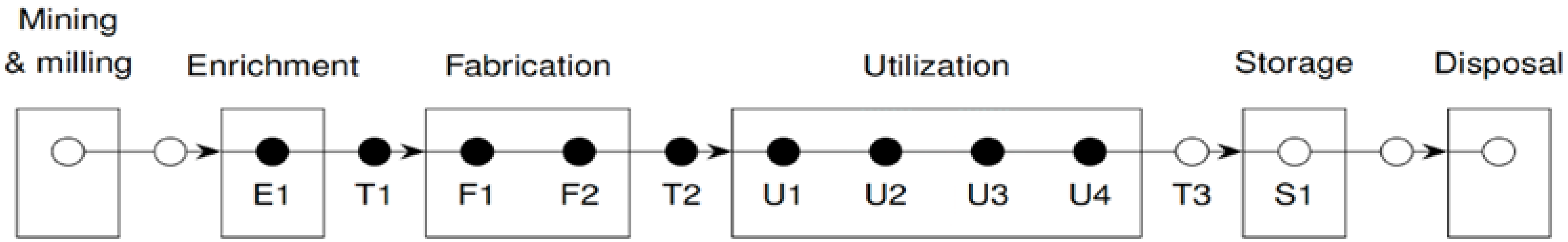


[Fuel Cycle]

Decay Heat

Criticality Safety

Shielding & Dose



E1 – UF_6 enrichment
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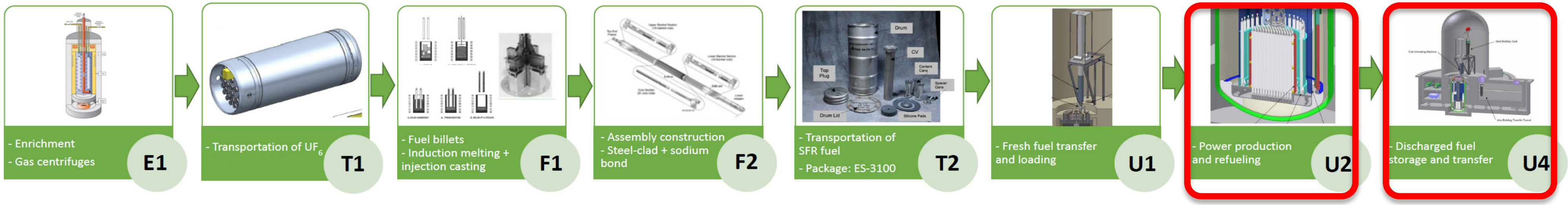
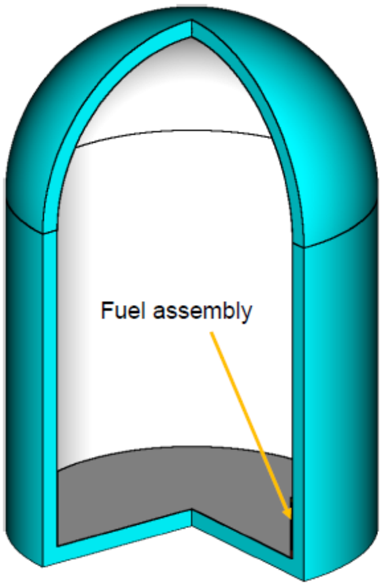


Shielding & Dose Analyses for Metallic Fuels / SFRs

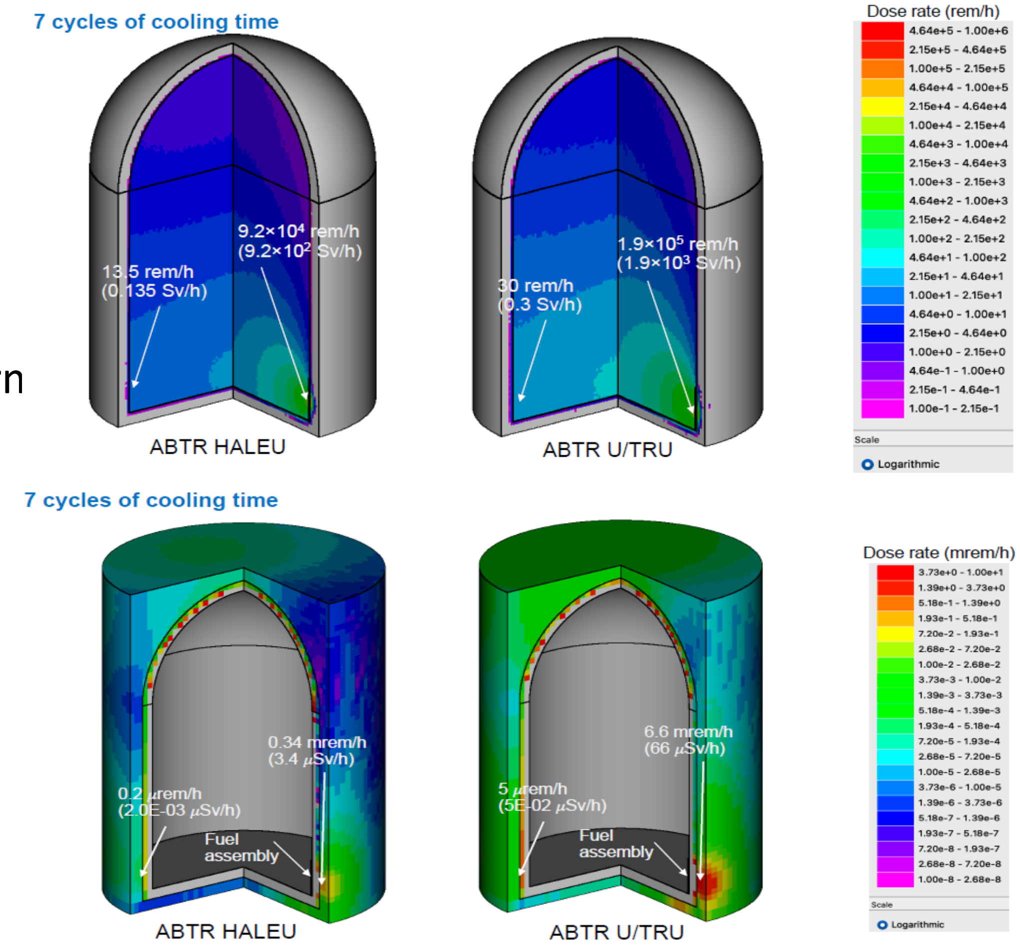
Scenario 1: Release of fission products during operation / refueling (U3)

- Accident: Seismic event causing the refueling machine to fall and release the fuel assembly.
- Analysis: Determine fuel inventory and perform SCALE radiation dose calculations.



3D view of the CB with front quarter segment removed

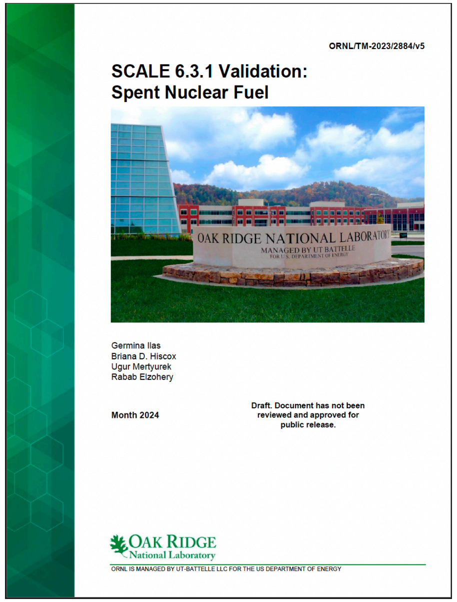
- Generated inventories used for radiative source tern
  - Leveraged from the non-LWR demonstration source term work
  - TRITON & ORIGAMI for inventories
  - MAVRIC for shielding & dose
- Radionuclide inventories used to support downstream analyses.
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses

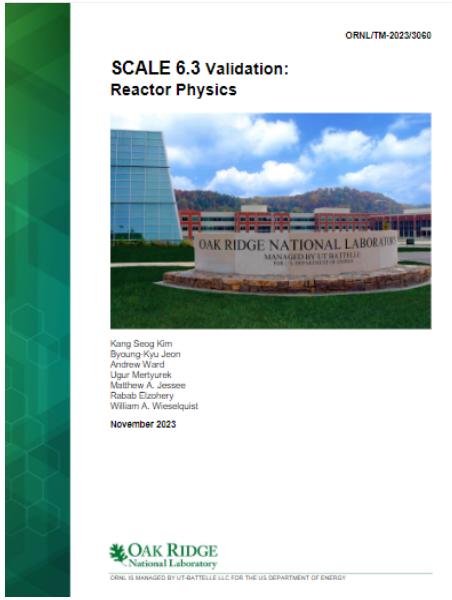


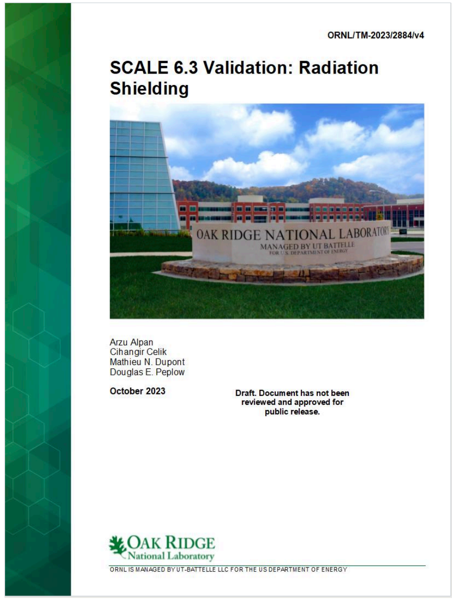
3D total dose rate maps of the containment building, generated from SCALE MAVRIC

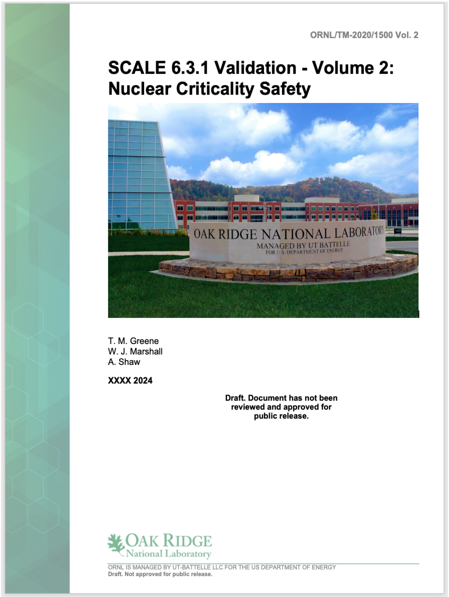
NRC's Computer Codes and Validation

SCALE Validation in Four Major Areas (Criticality Safety, Radiation Shielding, Reactor Physics, and Spent Fuel Inventory)









SFRs

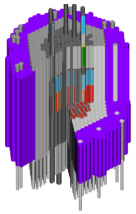


Figure 5.7. EBR-II SCALE model [62].

Table 5.4. Eigenvalue results for the high-fidelity EBR-II benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [7]	1.00927	± 0.00618	reference
SCALE 6.3.1/KENO-VI CE ENDF/B-VII.1	1.00722	± 0.00010	-205 (± 618)
SCALE 6.3.1/KENO-VI CE ENDF/B-VIII.0	1.00691	± 0.00013	-236 (± 618)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

HTGRs

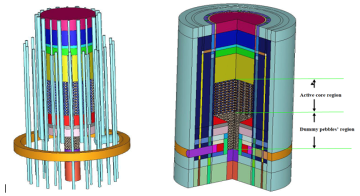


Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

Table 5.1. Eigenvalue results for high-fidelity HTR-10 benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value [6]	1.00000	0.00370	reference
SCALE/KENO-VI CE ENDF/B-VII.1	1.00303 ± 0.00041	0.99661 ± 0.00031	303 ± 370
SCALE/KENO-VI CE ENDF/B-VIII.0	1.00604 ± 0.00027	0.99919 ± 0.00026	604 ± 370
SCALE/KENO-VI 252-group ENDF/B-VII.1	1.00265 ± 0.00031	0.99595 ± 0.00025	265 ± 370
SCALE/KENO-VI 252-group ENDF/B-VIII.0	1.00376 ± 0.00027	0.99746 ± 0.00025	376 ± 370

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

MSRs

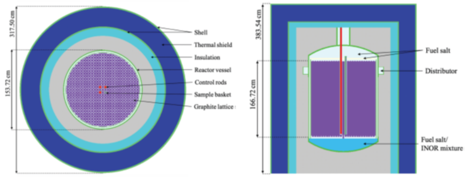


Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

Table 5.3. Eigenvalue results for the high-fidelity MSRE benchmark.

	k_{eff}	σ	Δk_{eff}^{a} (pcm)
Benchmark value	0.99978	± 0.00420	reference
SCALE 6.3.1/Shift CE ENDF/B-VII.1	1.019016	± 0.00010	1924 (± 420)
SCALE 6.3.1/Shift CE ENDF/B-VIII.0	1.021833	± 0.00010	2205 (± 420)

[a] Calculated as 10^5(k-eff$_{calculated}$ – k-eff$_{benchmark}$).

SCALE has been heavily validated for standard fuel designs in LWRs. SCALE 6.3 validation efforts are underway to validate SCALE for several advanced non-LWR systems.

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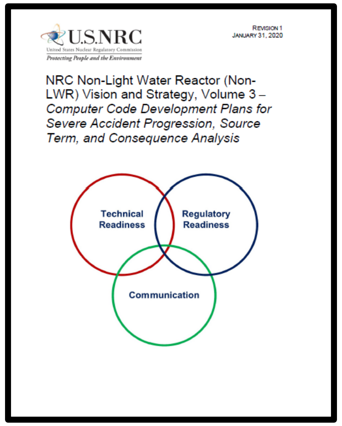
Applications of non-LWR Demonstration Project - Kairos Hermes Construction Permit Application Support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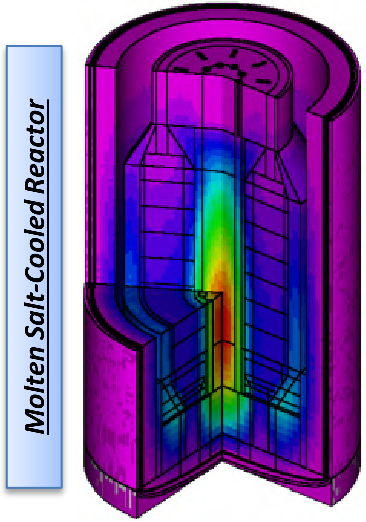
[Source Term]



UCB Mk1 PB-FHR

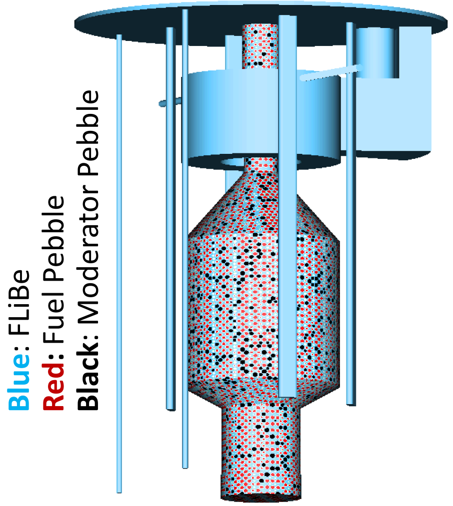
- 236 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling

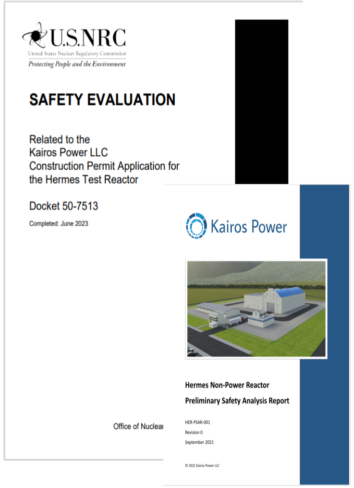
- *Generated a library of well-tested & demonstrated non-LWR reference plant models in SCALE & MELCOR.*
- *Models can be heavily leveraged to support licensing reviews.*

- *Leveraged the FHR model to supportrt the licensing review of Hermes I*
  - *Similarities between the UCB Mk1 & Hermes I noted*
  - *Leveraged existing models & insights from non-LWR demonstration project*
- *SCALE and MELCOR used for analyzing various scenarios (e.g., loss of forced circulation, accidental control rod withdrawal)*

Kairos Hermes I

- 35 MWth reactor at atmospheric pressures
- Flibe cooled & Pebble fueled (TRISO) at 19.75 wt.% U-235
- Online refueling





Non-LWR demonstration project was instrumental in an effective and efficient review of a first of a kind non-LWR.

For More Information

SCALE/MELCOR non-LWR source term demonstration project	

• Heat-pipe reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	June 29, 2021

• High-temperature gas-cooled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	July 20, 2021

• Fluoride-salt-cooled high-temperature reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 14, 2021

• Molten-salt-fueled reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 13, 2022

• Sodium-cooled fast reactor workshop
  ◦ Slides
  ◦ Video Recording EXIT
• SCALE report
• MELCOR report
	September 20, 2022

SCALE/MELCOR non-LWR fuel cycle demonstration project	

• High-temperature gas-cooled reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording EXIT
  ◦ SCALE Report
  ◦ MELCOR Report
	February 28, 2023

• Sodium-cooled fast reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	September 20, 2023

• Molten salt reactor fuel cycle workshop
  ◦ Slides
  ◦ Video Recording
  ◦ SCALE Report
  ◦ MELCOR Report
	July 11, 2024

• Microreactor fuel cycle workshop	Coming in 2025
• Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration ◦ Report	December 15, 2023

Public workshop videos, slides, reports at advanced reactor source term webpage