Storage and Transportation of TRISO and Metal Spent Nuclear Fuels

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

Andy Bielen, Ph.D.

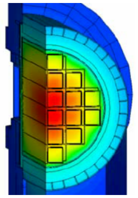
Office of Nuclear Regulatory Research
Division of Systems Analysis
Fuel & Source Term Code Development Branch

Objectives

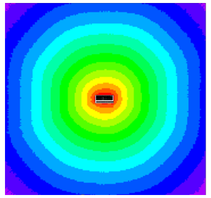
- *NRC's simulation capabilities supporting nuclear fuel safety for TRISO-particle fuel designs*
  - *Decay Heat*
  - *Neutron Multiplication & Criticality*
  - *Shielding and Radiation Protection*

- *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

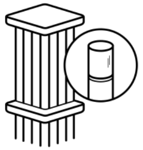
Computer codes used to determine:

- 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}

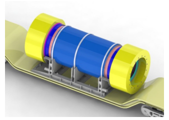
Codes must be validated against measured irradiated fuel data

10 CFR 20 – Radiation Protection

10 CFR 50/52 – Power Plants



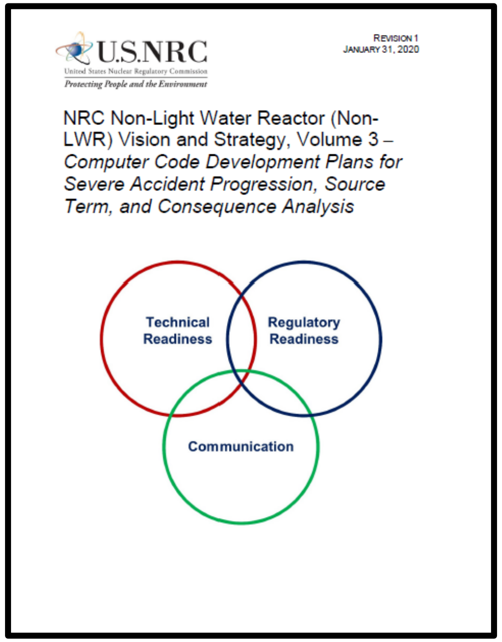
10 CFR 70 – Fuel Cycle Facilities



10 CFR 71 – Transportation

10 CFR 72 – SNF Storage

Non-LWR Source Term & Fuel Cycle Demonstration Projects



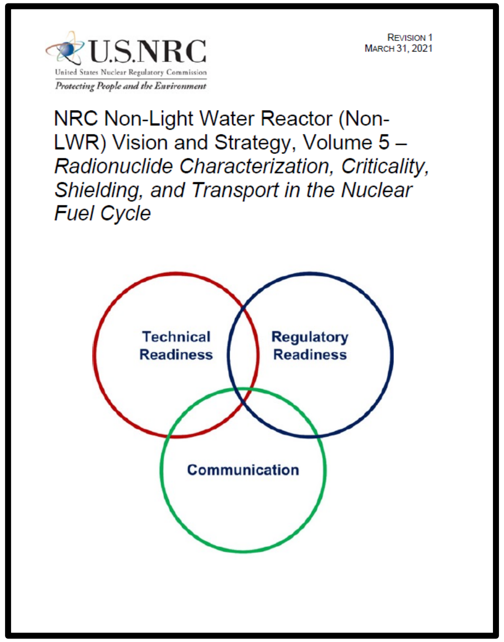
Source Term

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

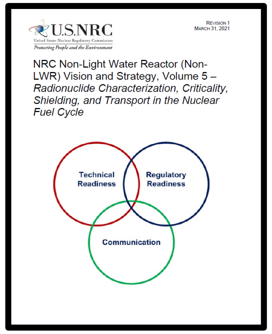
- Accident progression
- Thermal-hydraulic response
- Core heat-up, degradation, and relocation
- Fission product release and transport behavior



Fuel Cycle

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

Decay Heat, Criticality Safety, and Radiation Shielding / Dose



[Fuel Cycle]

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

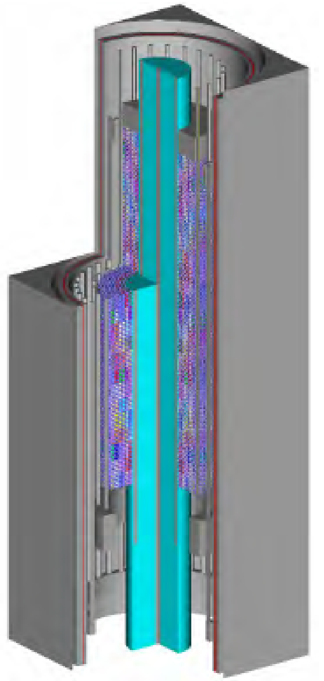
- SCALE/CSAS is used to perform criticality safety analyses. CSAS is a sequence that uses Monte Carlo transport codes KENO or Shift.
- Used to determine the multiplication factor of any system.

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

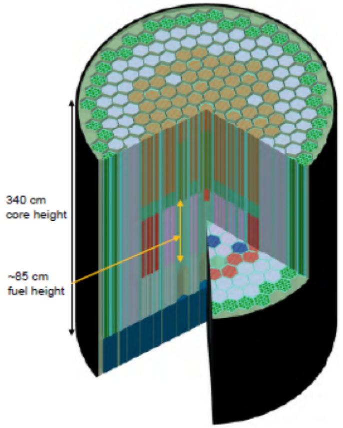
High-Temp. Gas Cooled Reactor



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)

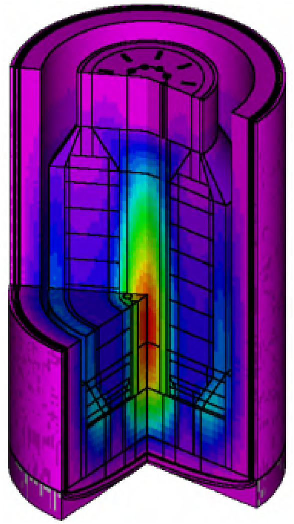
Sodium-Cooled Fast Reactor



ABTR

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

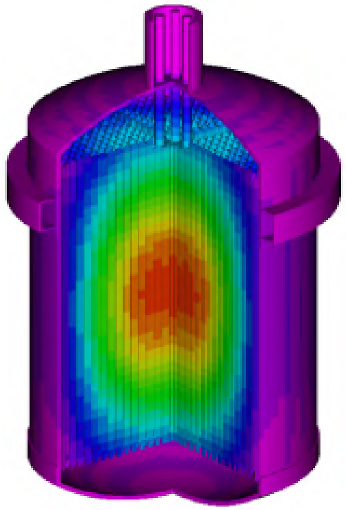
Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

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

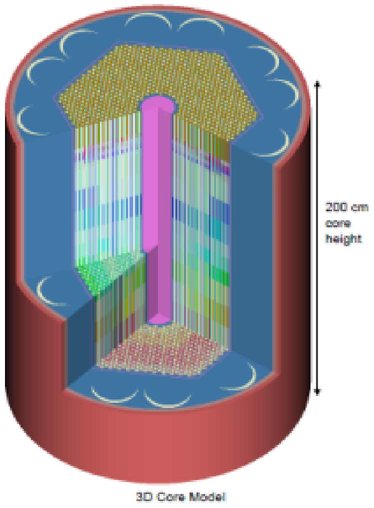
Molten Salt-Fueled Reactor



MSRE

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

Heat Pipe Reactor

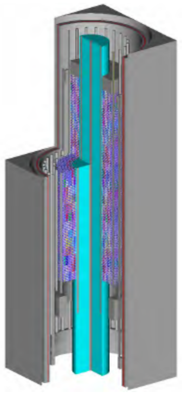


INL Design A

- 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

Pebble Bed Reactor Workflows

High-Temp. Gas Cooled Reactor



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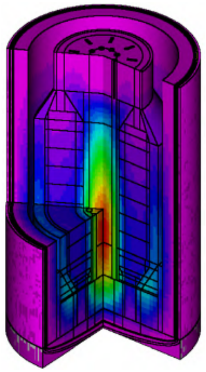

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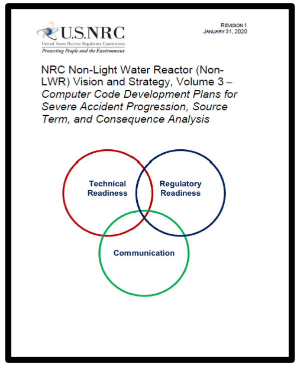
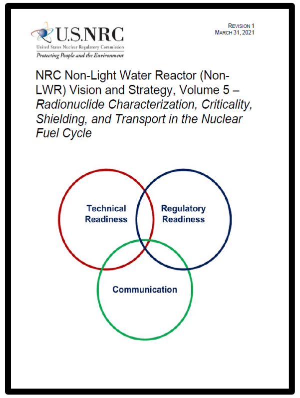

U.S.NRC

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


U.S.NRC

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*

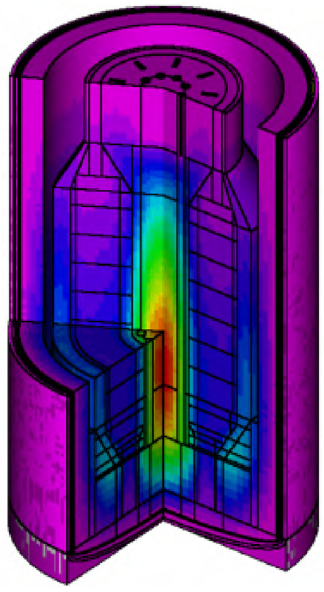
Fuel Cycle

Fuel Depletion, Decay Heat, & Radionuclide Inventory Generation

- Decay Heat
- Criticality Safety
- Shielding & Dose

Fuel Depletion, Decay Heat, and Nuclide Inventory Generation

Molten Salt-Cooled Reactor



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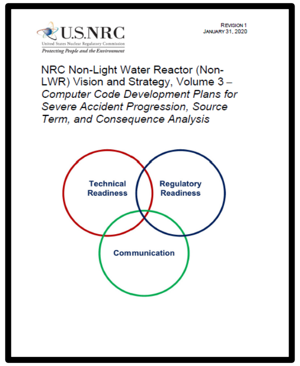

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Protecting People and the Environment

Revision 1
January 31, 2020

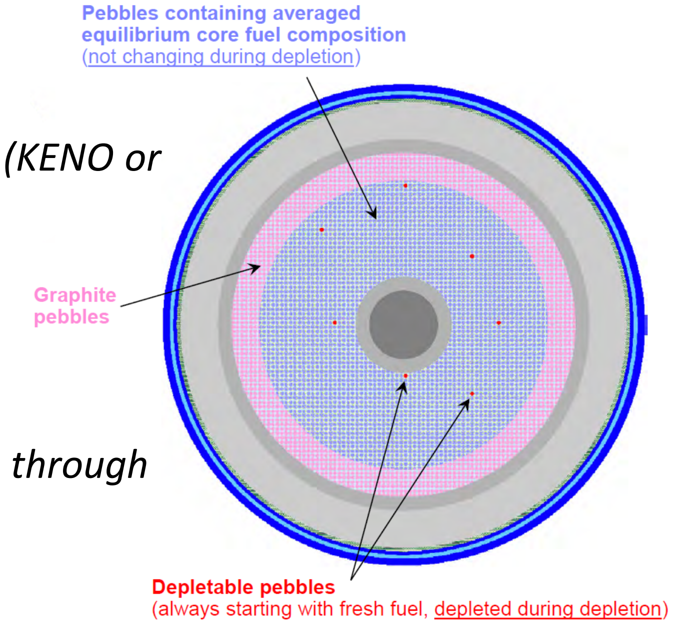
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 3 – *Computer Code Development Plans for Severe Accident Progression, Source Term, and Consequence Analysis*

Technical Readiness
Regulatory Readiness
Communication

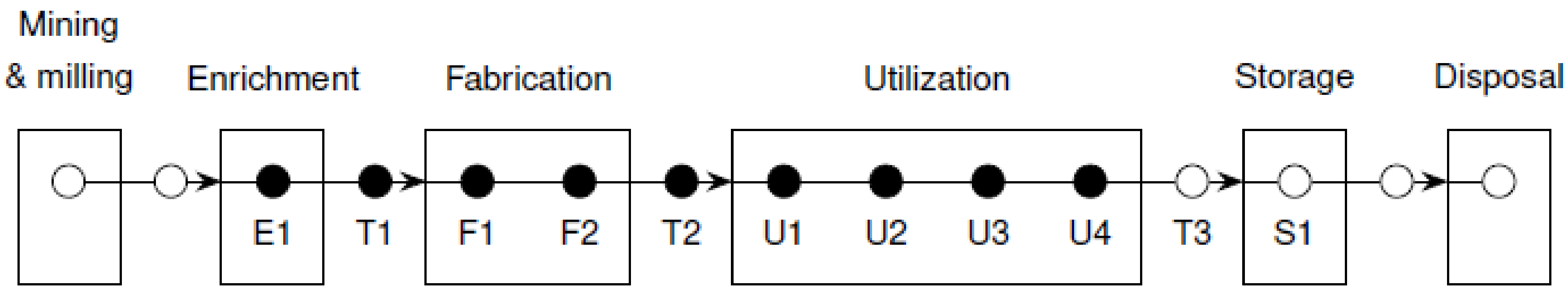
Fuel Depletion, Decay Heat, & Radionuclide Inventory Generation

Source Term

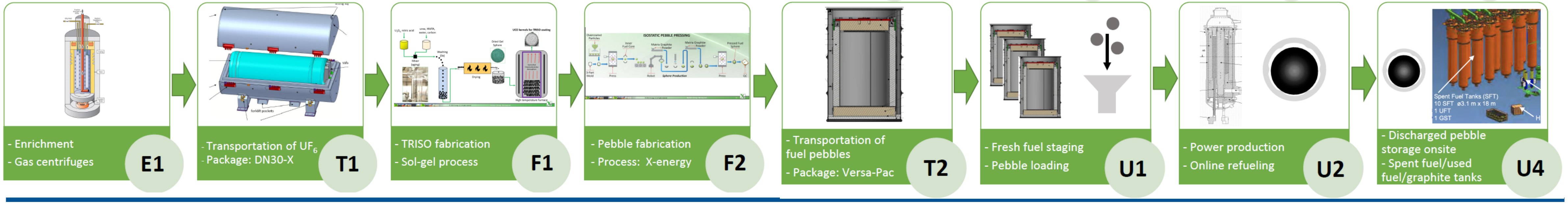
- SCALE/TRITON used for fuel depletion
  - *Continuous energy Monte Carlo physics or MG methods available (KENO or Shift)*
  - *MG methods utilize SCALE's double-het methods*
  - *Equilibrium inventories generated via SLICE method*
    - *SCALE Leap-In Method for Cores at Equilibrium*
    - *Generates region-average fuel inventories*
    - *Accounts for average behavior of pebbles as they transverse through the core*
- *Radionuclide inventories used to support downstream analyses.*
  - *MELCOR for severe accident progression & radionuclide transport*
  - *ORIGAMI for decay heat analyses; utilizes the ORIGEN libraries from TRITON*



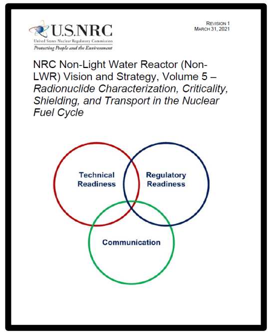
High Temperature Gas-Cooled Reactors Fuel Cycle



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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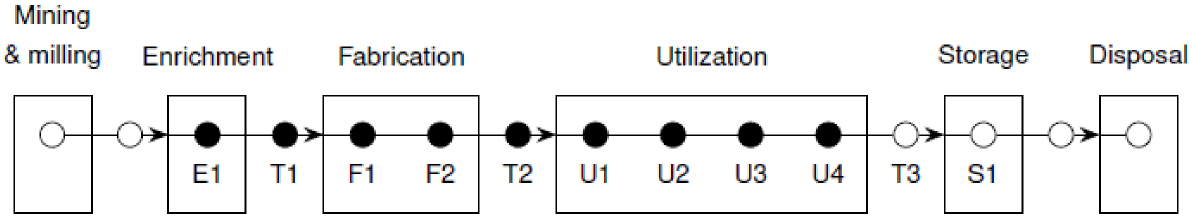
Revision 1
March 31, 2021
NRC Non-Light Water Reactor (Non-LWR) Vision and Strategy, Volume 5 – *Radionuclide Characterization, Criticality, Shielding, and Transport in the Nuclear Fuel Cycle*

Fuel Cycle

Decay Heat

Criticality Safety

Shielding & Dose



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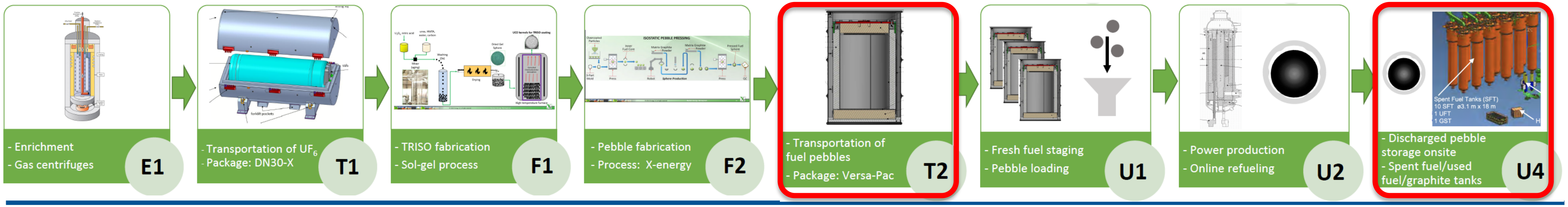
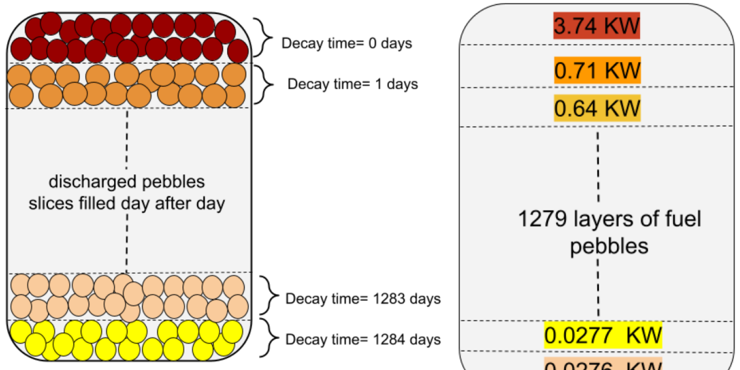
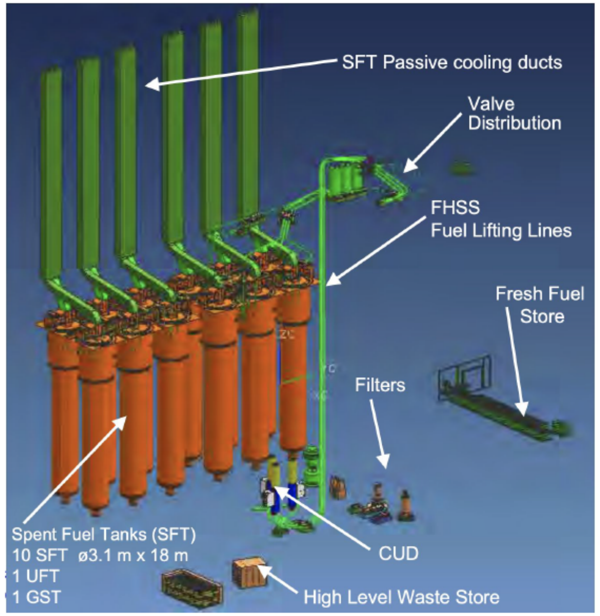


Decay Heat Analyses for TRISO-based Fuels

- *Vehicle / collision strike with a spent nuclear fuel storage tank loaded with spent TRISO-pebbles.*
  - *Once burnup limits are reached, pebble is moved into a spent fuel tank, with a capacity of holding ~620K pebbles.*
  - *Discharge rate – 483 pebbles / day; 1,284 days to fill spent fuel tank.*



Spent fuel tank



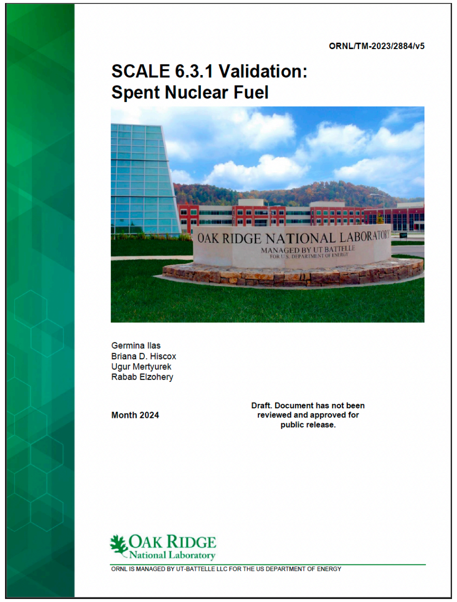
PBMR-400 FHSS with pebble storage tanks*

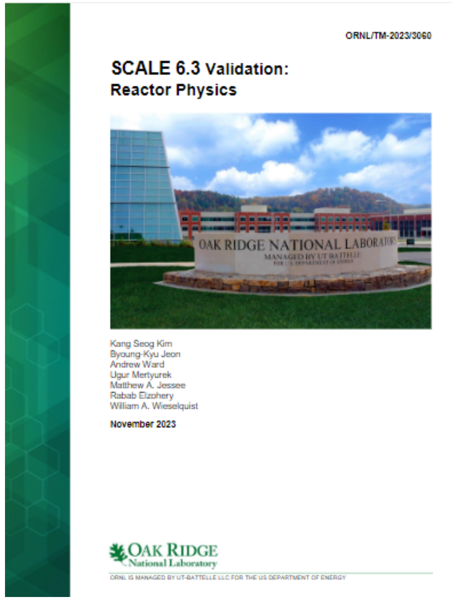
- *Determine average spent fuel pebble inventory after discharge*
  - Leveraged from the non-LWR demonstration source term work (for HTGR)
  - TRITON & ORIGAMI used for generating inventories & performing decay-correction from ORIGEN reactor libraries

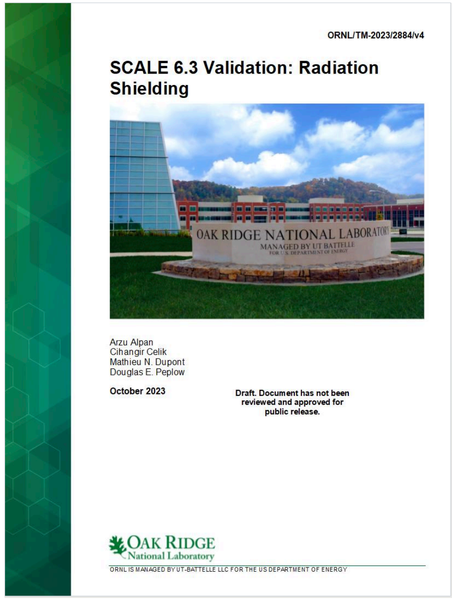
- *Radionuclide inventories used to support downstream analyses.*
  - MELCOR for severe accident progression & radionuclide transport
  - MAVRIC for shielding & dose analyses

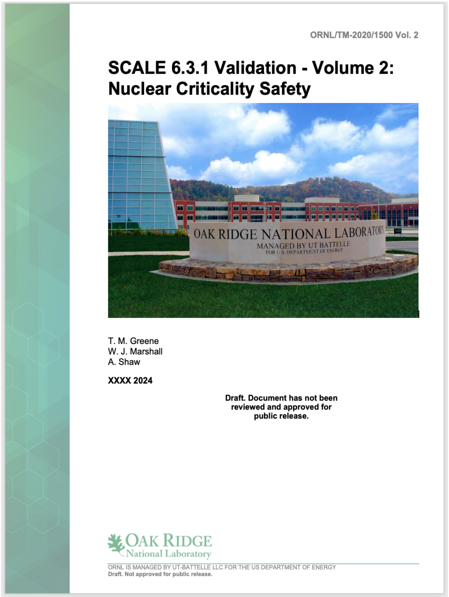
NRC's Computer Codes and Validation

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









SFRs

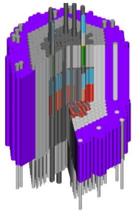


Figure 5.7. EBR-II SCALE model [62].

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

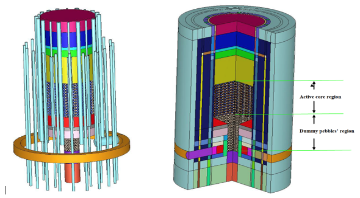


Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

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

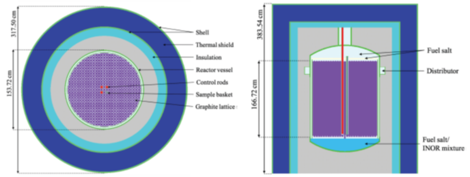


Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

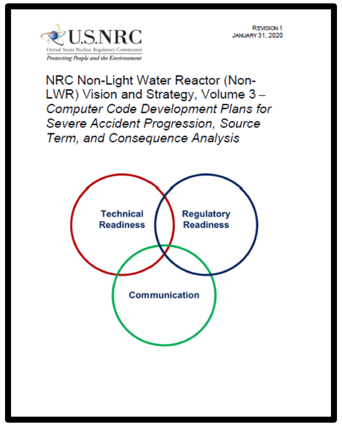
Table 5.3. Eigenvalue results for the high-fidelity MSRE benchmark.

	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.

Applications of non-LWR Demonstration Project - Kairos Hermes Construction Permit Application Support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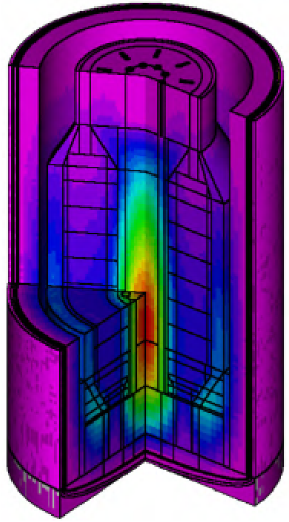
REVISION 1
JANUARY 31, 2020

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]()

Molten Salt-Cooled Reactor



UCB Mk1 PB-FHR

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

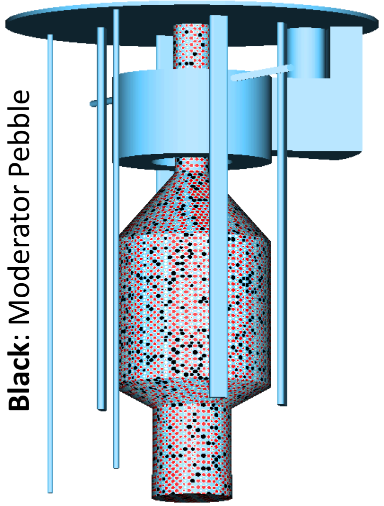
- *Generated a library of well-tested & demonstrated non-LWR reference plant models in SCALE & MELCOR.*
- *Models can be heavily leveraged to support licensing reviews.*

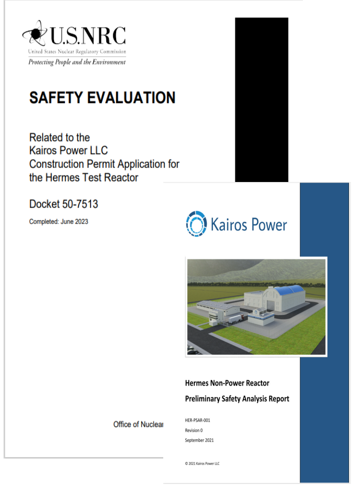
- *Leveraged the FHR model to supportrt the licensing review of Hermes I*
  - *Similarities between the UCB Mk1 & Hermes I noted*
  - *Leveraged existing models & insights from non-LWR demonstration project*
- *SCALE and MELCOR used for analyzing various scenarios (e.g., loss of forced circulation, accidental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

Blue: FLiBe
Red: Fuel Pebble
Black: Moderator Pebble

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SAFETY EVALUATION

Related to the
Kairos Power LLC
Construction Permit Application for
the Hermes Test Reactor

Docket 50-7513

Completed: June 2023

Office of Nuclear

Kairos Power

Hermes Non-Power Reactor
Preliminary Safety Analysis Report

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