Overview of MELCOR/SCALE non-LWR Source Term and Fuel Cycle

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Volume 3

Non-LWR Severe Accident and Source Term Analysis

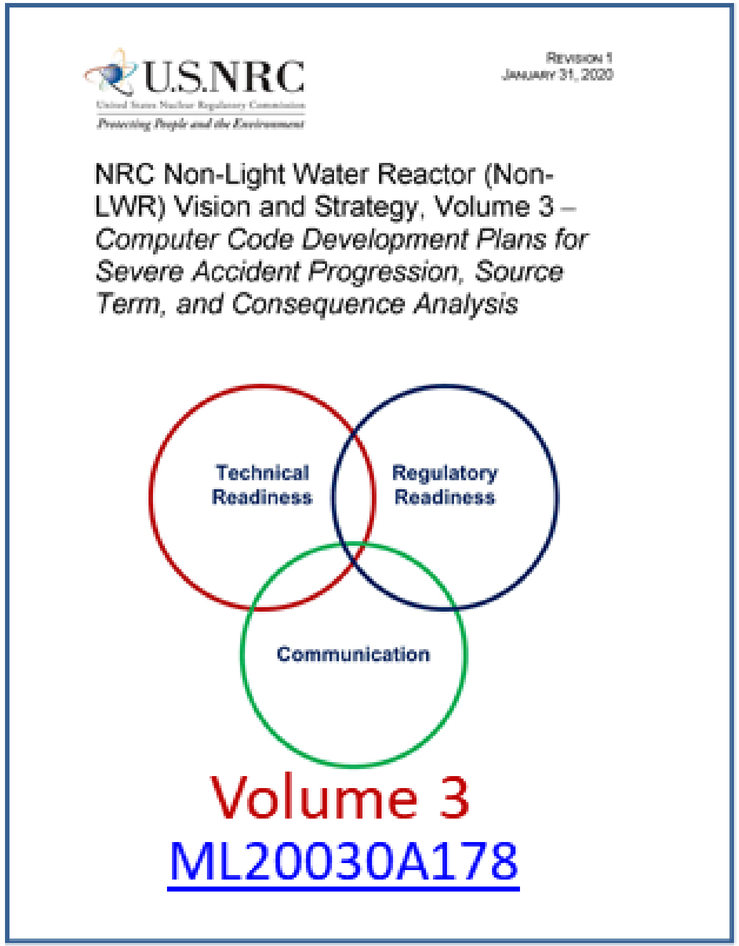
United States Nuclear Regulatory Commission
Protecting People and the Environment

Severe Accident Analysis: Objectives

- Understand severe accident progression in non-LWRs
  - Provide insights for regulatory guidance
  - Build staff knowledge and expertise in modeling non-LWRs

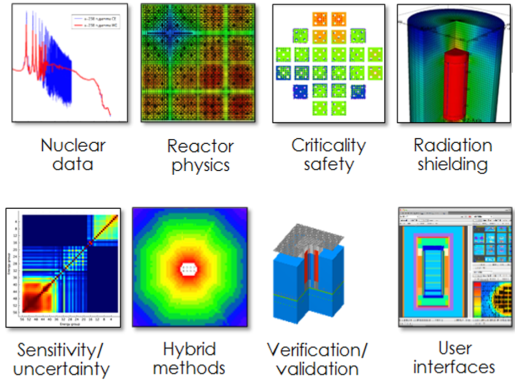
- Facilitate dialogue on staff's approach for source term

- Ensure tool & model readiness for licensing non-LWRs
  - Develop necessary modeling capabilities in SCALE & MELCOR
  - Identify accident characteristics and uncertainties affecting source term

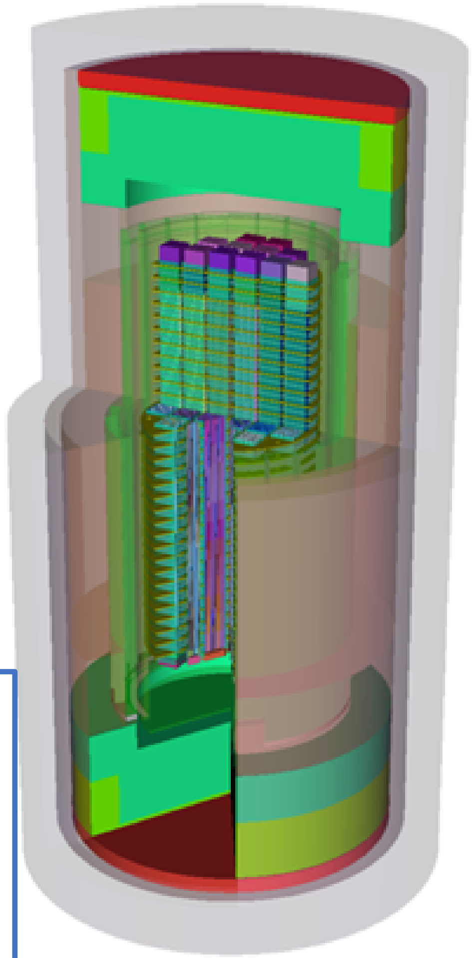


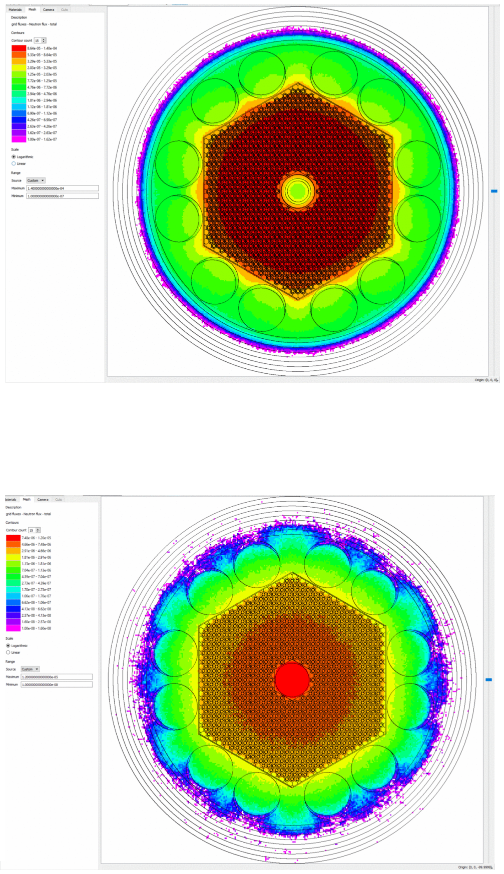
Severe Accident Analysis: SCALE Code

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

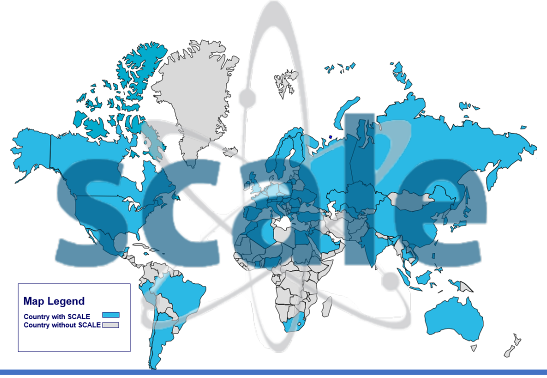


How Is It Used?
SCALE is used to support licensing activities (e.g., analysis of spent fuel pool criticality, generating reactor physics and decay heat parameters for design-basis accident analysis, and review of consolidated interim storage facilities, burnup credit).





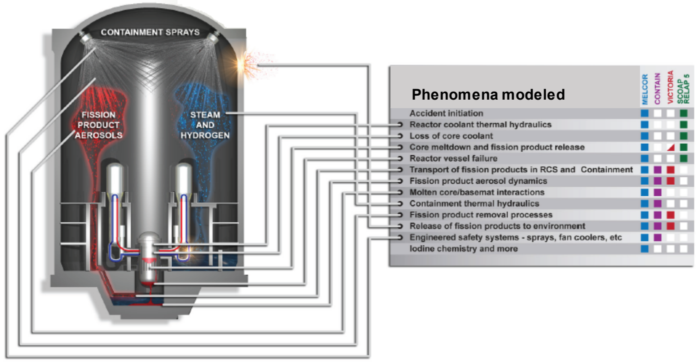
Who Uses It?
SCALE is used by the NRC and in 61 countries (about 11,000 users and 33 regulatory bodies).

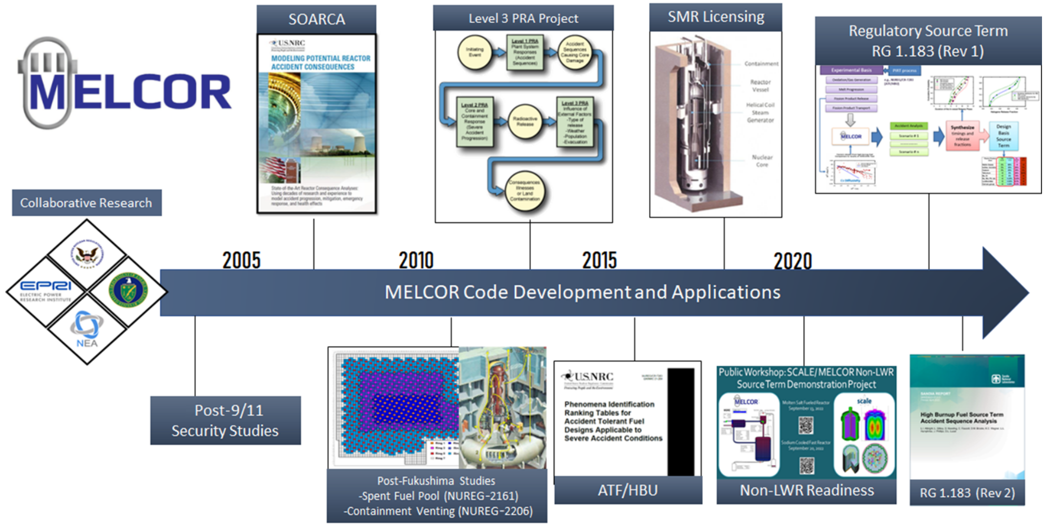


How Has It Been Assessed?
SCALE has been validated against numerous critical experiments that cover a range of fuel and moderator materials and geometries, and against measured PWR and BWR spent fuel isotopic composition and decay heat measurements.

Severe Accident Analysis: MELCOR Code

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





Phébus-Fission Products & Source Term Program	Behavior of Iodine Project (BIP)	Experimental Program for Iodine Chemistry Under Radiation (EPICUR)	Source Term Evaluation and Mitigation (STEM) Project	Benchmark Study of the Accident at Fukushima (BSAF) Project	Management and Uncertainties of Severe Accidents (MUSA)	Experiments on Source Term for delayed Releases (ESTER) Reduction of Severe Accident Uncertainties (ROSAU)	Thermodynamic Characterization Of Fuel debris and Fission (TCOFF-2)	Fukushima Accident Information Collection & Evaluation (FACE)
1988-2010	2006-2019	2005-2016	2011-2019	2013-2018	2019-2023	2020-2024	2022-2024	2023-2026

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

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

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

Severe Accident Analysis: Approach

1. Build representative SCALE core models and MELCOR full-plant models

2. Select scenarios that demonstrate code capabilities for key phenomena

3. Perform simulations
   - SCALE - generate decay heat, core radionuclide inventory, and reactivity feedbacks
   - MELCOR - model accident progression, plant response, and source term

Severe Accident Analysis: Project Scope

- Five Types of Non-LWRs Analyzed for Source Term Demonstration

- *2021*
  - Heat Pipe Reactor – INL Design A
  - High-Temperature Gas-cooled Pebble-bed Reactor – PBMR-400
  - Molten-salt-cooled Pebble-bed Reactor – UCB Mark 1

- *2022*
  - Molten-salt-fueled Reactor – MSRE
  - Sodium-cooled Fast Reactor – ABTR

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

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

SCALE input models available here.

MELCOR input models available upon request.

Severe Accident Analysis: Molten-salt-cooled Pebble-bed Rx – UCB Mark 1

Reactor Characteristics

- 236 MWth reactor
- Atmospheric pressures
- Flibe cooled
- Pebble fueled (TRISO) at 19.9 wt.% U-235
- Online refueling
- Direct Reactor Auxiliary Cooling System (DRACS)
  - 3 trains –2.36 MW/train
  - Each train has 4 loops in series
  - Primary coolant circulates to DRACS heat exchanger
  - Molten-salt loop circulates to the thermosyphon-cooled heat exchangers (TCHX)
  - Water circulates adjacent to the secondary salt tube loop in the TCHX


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
Jae Keun Choi
Alexandre Y.K. Chong
Massimiliano Fratoni
Sea Hong
Lakshana R. Huddar
Kathryn D. Huff
David L. Krumwiede
Michael R. Laufer
Madicken Munk
Raluca O. Scarlat
Nicolas Zweibaum
Ehud Greenspan
Per F. Peterson

UCBTH-14-002

September 30, 2014
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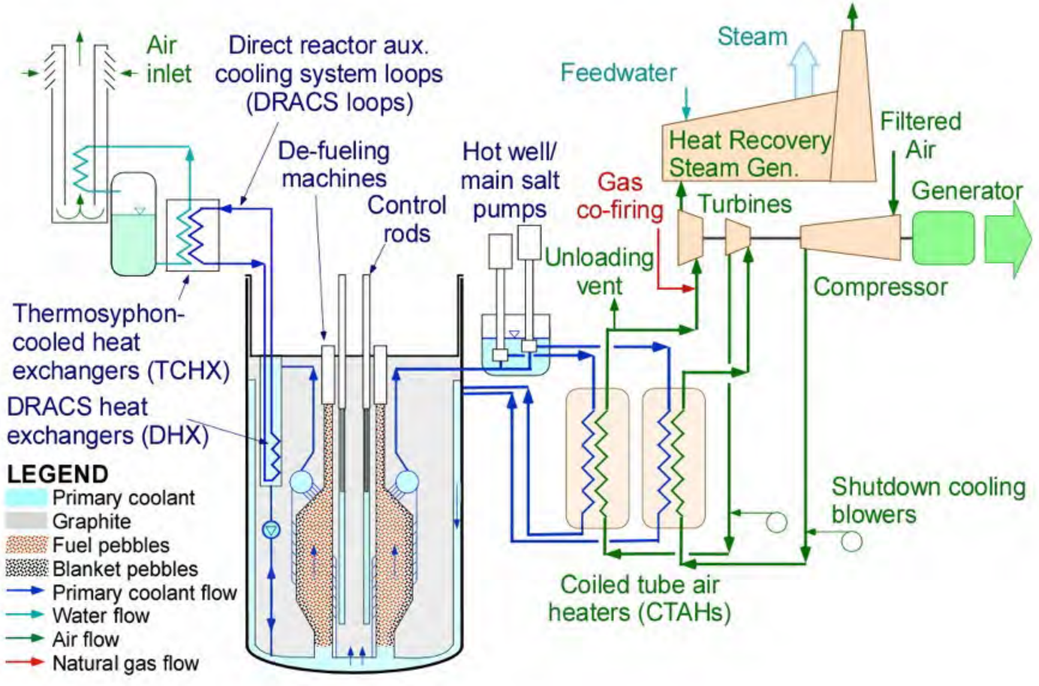
University of California, Berkeley

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

U.S. Department of Energy

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



Accidents Modeled

- ATWS –Anticipated transient without SCRAM
- SBO –Station blackout
- LOCA –Loss-of-coolant accident

Severe Accident Analysis:
Molten-salt-cooled Pebble-bed Rx – UCB Mark 1

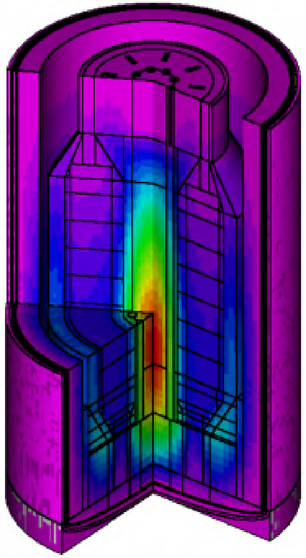
Code Improvements

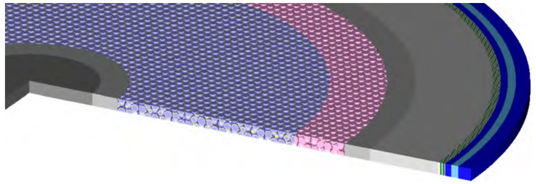
scale

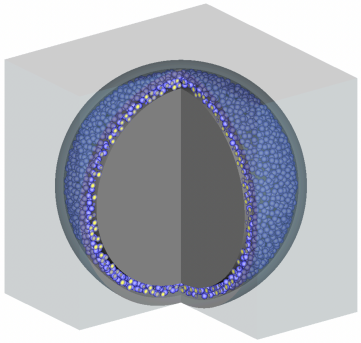
- New interface for rapid depletion of TRISO fuel for low computational costs (*increased efficiencies for performing wide array of sensitivity studies)*
- Developed workflow for pebble-bed reactor equilibrium core generation using SCALE’s efficient multigroup treatment for double heterogeneous systems

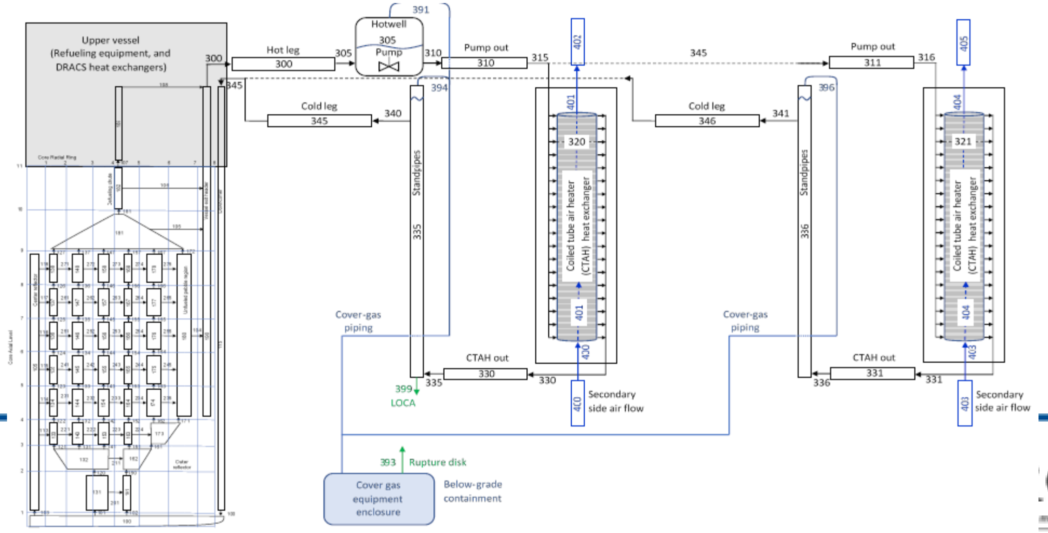
MELCOR

- Added a generic equation of state utility for thermal hydraulic analysis in advanced reactor working fluids
- Fission product transport and retention models added for molten salts
- Improved fission product release models for TRISO
- Point-kinetic enhancements for reactivity insertion









Severe Accident Analysis:
Molten-salt-cooled Pebble-bed Rx – UCB Mark 1

<u>ATWS</u>

- Fuel heat-up was limited by reactivity feedback and the passive decay heat removal system

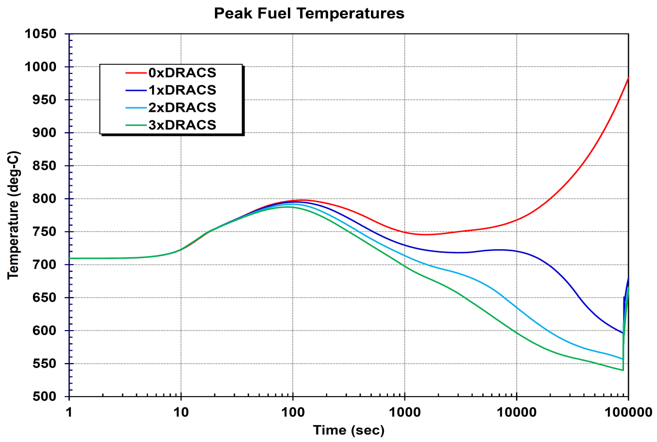
<u>SBO</u>

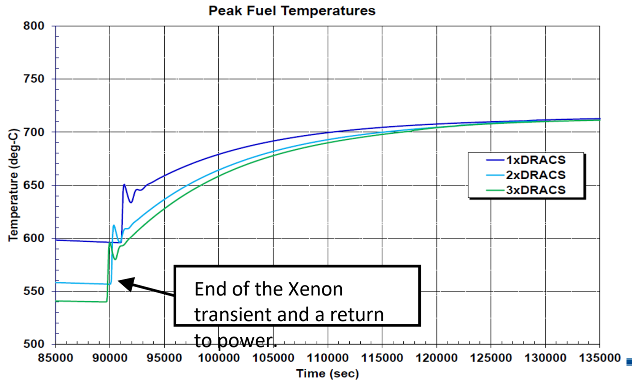
- With failure of the passive decay heat removal system, coolant boiling occurred over the course of several days

<u>LOCA</u>

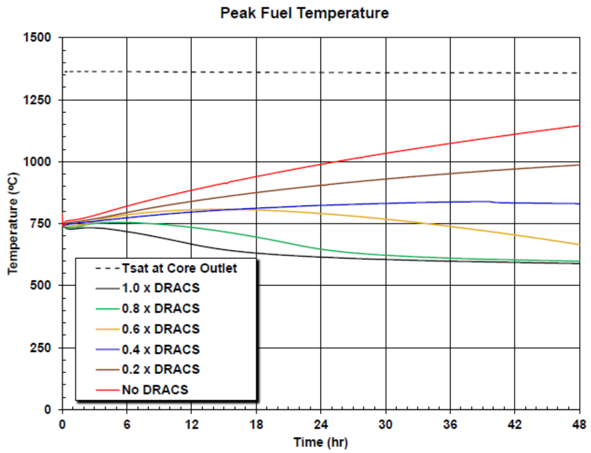
- With one train of decay removal system operating, coolant boiling was possibly averted.
- With failure of the passive decay heat removal system, fuel damage occurred.

ATWS with variable DRACS

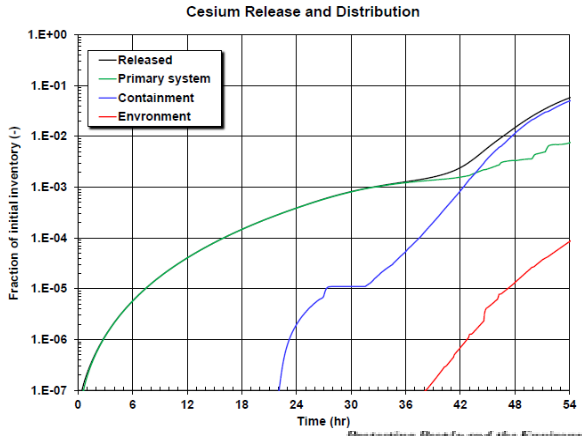




SBO



LOCA

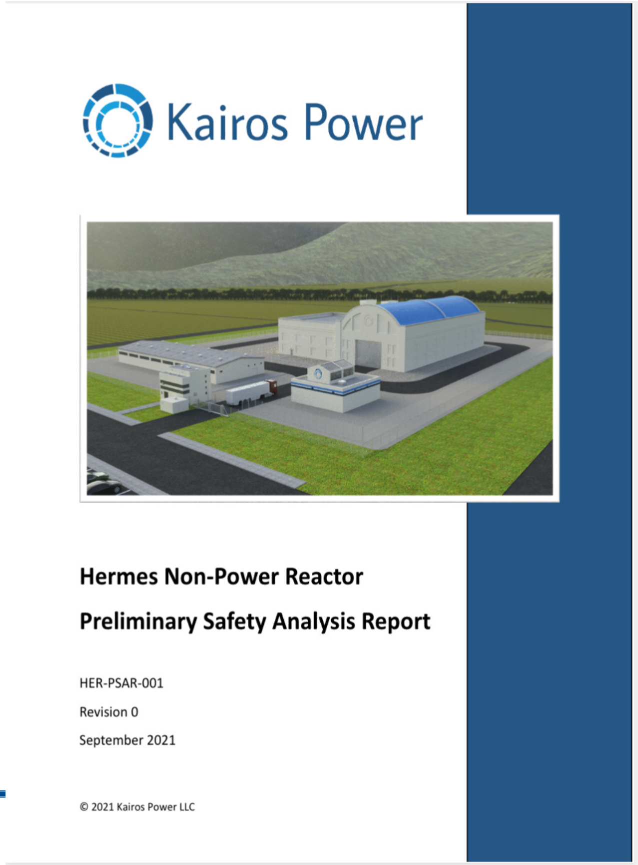


Severe Accident Analysis:
Hermes I Construction Permit Application

- On September 29, 2021, Kairos Power, LLC (KP) submitted a construction permit application to the NRC, requesting approval for their Hermes 35 MWth, non-power reactor facility.

- Leverage the UCB-Mark 1 FHR plant model to support Hermes analysis (January-March 2022). Scope was limited to design-basis events (i.e., no fuel uncovery).

- Provided NRR with SCALE and MELCOR analyses that supported their review looking at:
  - reactor heat-up scenario (e.g., loss of forced circulation),
  - insertion of excess reactivity scenario (e.g., accidental control rod withdrawal)


Kairos Power

Hermes Non-Power Reactor

Preliminary Safety Analysis Report

HER-PSAR-001

Revision 0

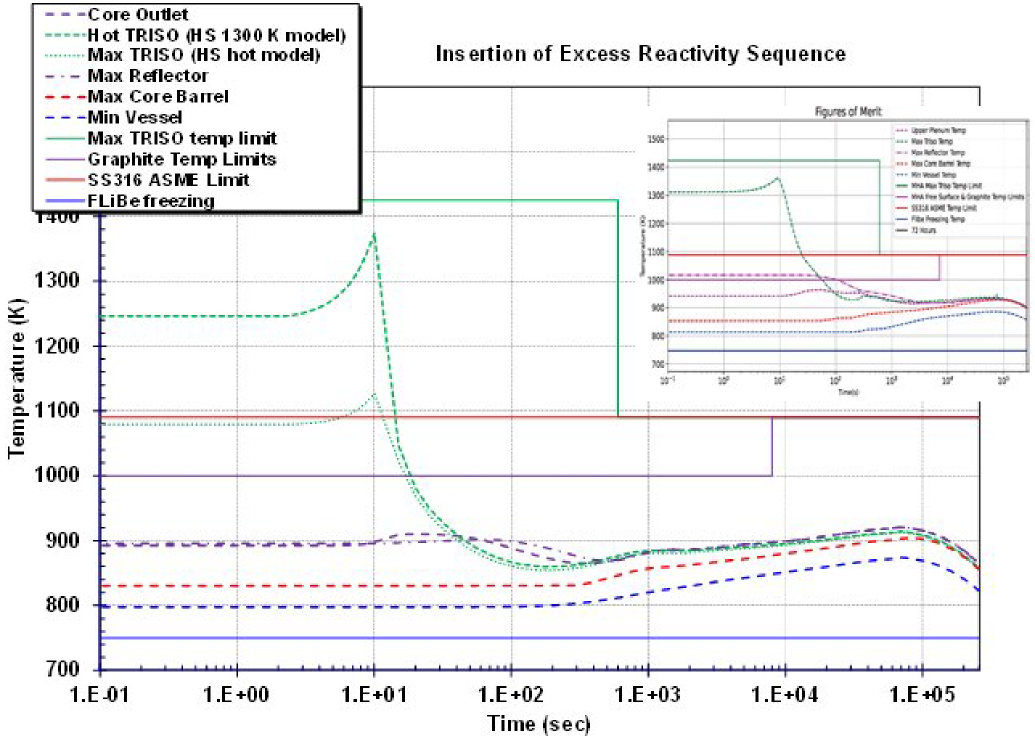
September 2021

© 2021 Kairos Power LLC

Severe Accident Analysis: Hermes I: SCALE/MELCOR Results

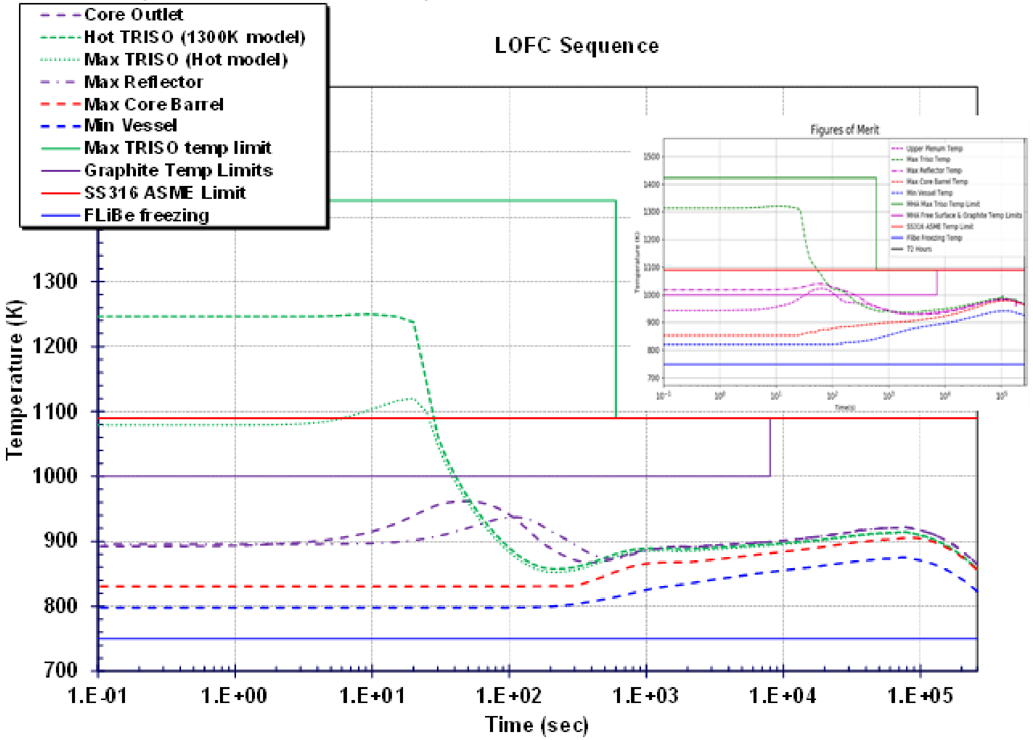
Insertion of Excess Reactivity

Withdrawal of control element inserts 3.02$ over 100 seconds
Reactor trips on high power



Loss of Forced Circulation

Concurrent trip of primary and intermediate coolant pumps
Reactor trips on overtemperature



MELCOR results as compared with PSAR (upper right)

Severe Accident Analysis:
Pebble-bed gas-cooled reactor – PBMR-400

<u>Reactor Characteristics</u>

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

<u>New Modeling Capabilities</u>

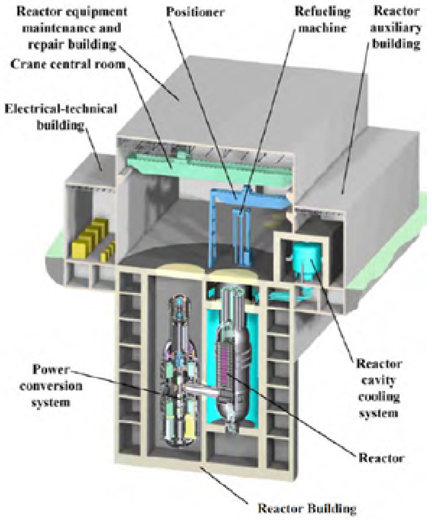
- SCALE: Interface for rapid depletion of TRISO fuel for efficient computational costs (*increased efficiencies for performing wide array of sensitivity studies)*
- MELCOR: TRISO fuel pebble thermal response, radionuclide diffusion, and failure models. Leveraged modeling efforts performed under NGNP (2006-2013)

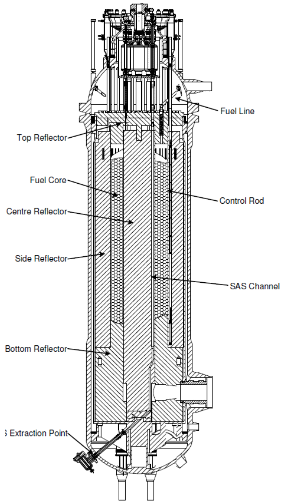
<u>Accidents Modeled</u>

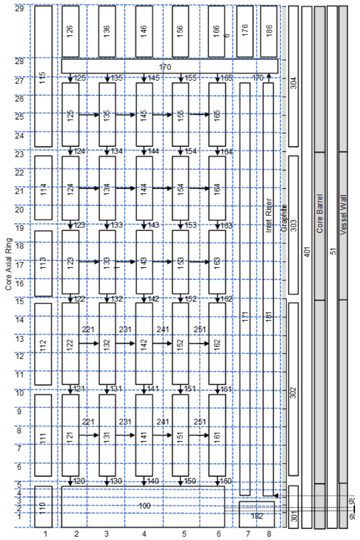
- Depressurized loss-of-forced circulation

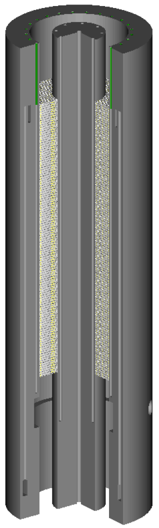
<u>Insights</u>

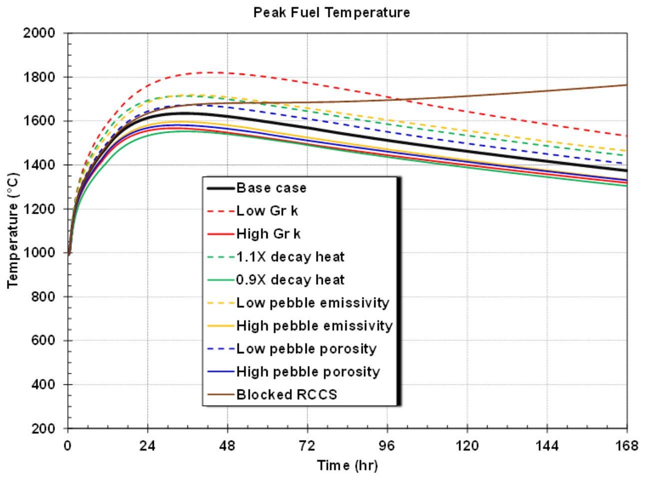
- Graphite oxidation from air ingress does not generate sufficient heat to impact fuel
- Passive heat dissipation into reactor cavity limits release from fuel failure
- A low graphite conductivity has the largest impact on the peak fuel temperature and release

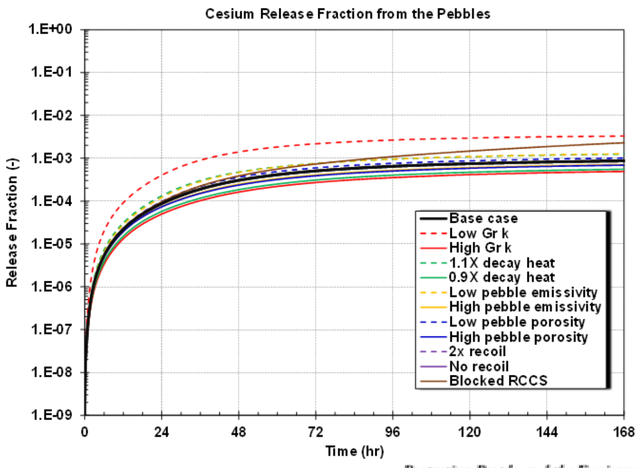












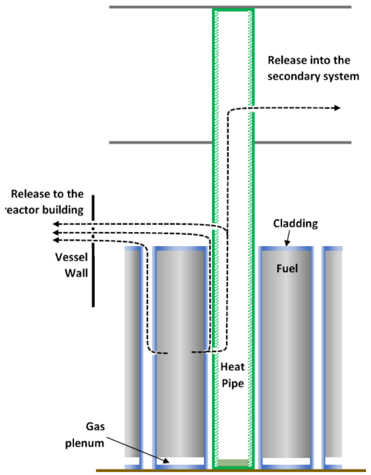
Severe Accident Analysis:
Heat pipe reactor – INL Design A

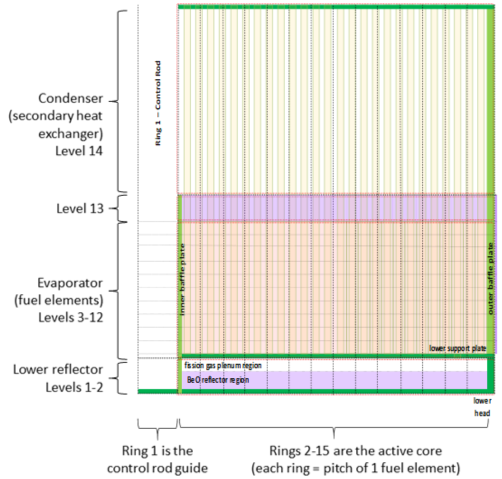
<u>Reactor Characteristics</u>
- 5 MWth with a 5-year operating lifetime
- 1,134 heat pipes fueled with metallic U (19.75 wt.% U-235)
- Reactivity controlled via control drums

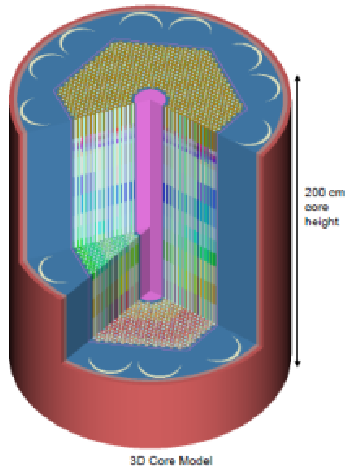
<u>New Modeling Capabilities</u>
- SCALE: New 302-group fast-spectrum library & 3D visualization improvements (*rapid model generations)*
- MELCOR: New thermophysical properties of sodium and potassium added, new HP-specific model (includes HP working fluid, HP connection to the secondary heat exchanger, and various HP failure modes)

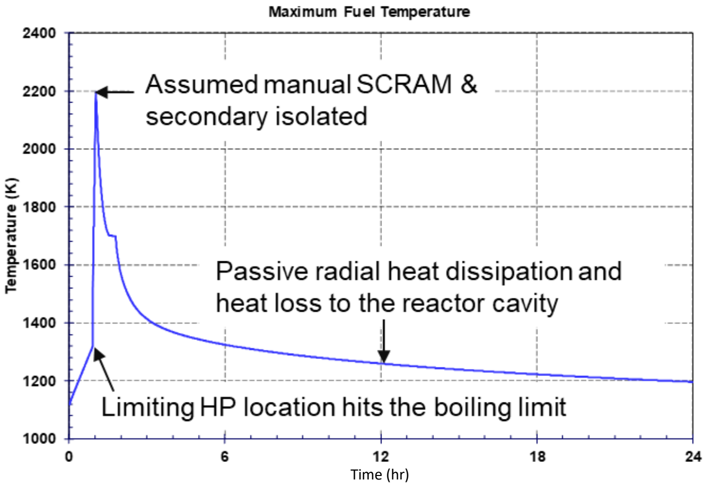
<u>Accidents Modeled</u>
- Transient overpower (TOP), loss-of-heat sink, and anticipated transient w/o SCRAM

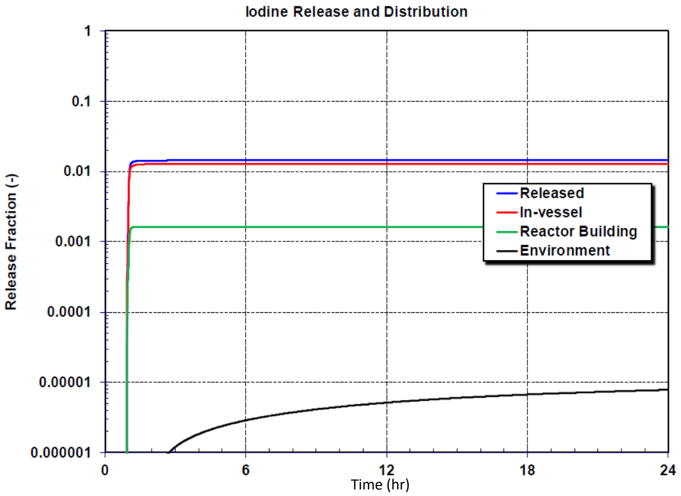
<u>Key Insights</u>
- Following SCRAM, passive heat dissipation into reactor cavity ends the release from fuel
- Heat pipe depressurization on failure drives the release from the reactor vessel into the reactor building
- Reactor building bypass requires two failures in a single heat pipe – one in the condenser region and another in the evaporator region











Severe Accident Analysis: Molten-salt-fueled reactor – MSRE

Reactor Characteristics

- 10 MWth reactor, graphite moderated at near atmospheric pressures
- Reactor fueled with dissolved fuel in molten salt (34.5 wt. % U-235)
- Fuel loop transit time ~25 seconds

New Modeling Capabilities

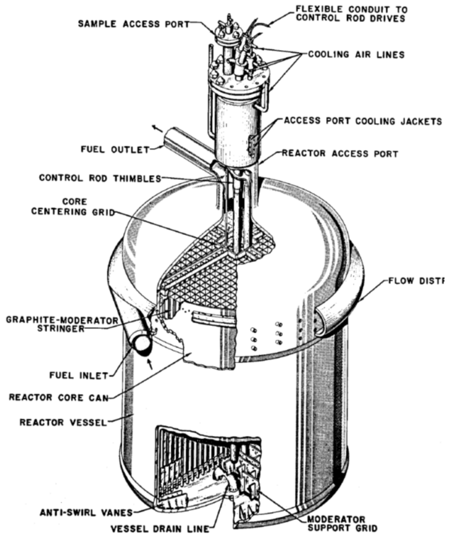
- SCALE: Modifications for handling liquid fuel, time-dependent system-average removal (e.g., simulating the off-gas system)
- MELCOR: Thermal hydraulic and equations of state for Flibe, Generalized Radionuclide Transport and Retention (GRTR) modeling framework, molten salt chemistry and physics pertaining to radionuclide transport, fluid fuel point kinetics

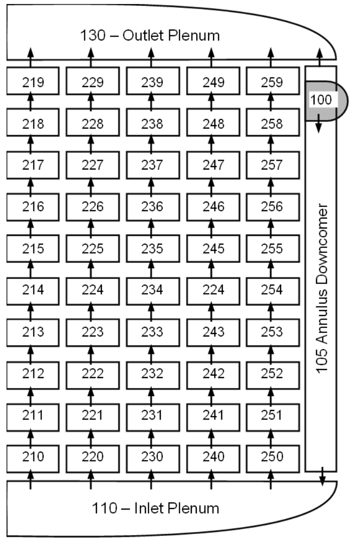
Accidents Modeled

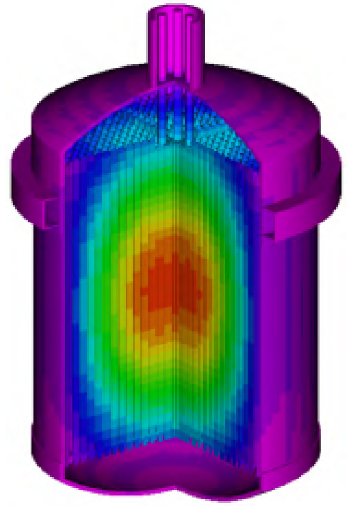
- Full reactor inventory molten salt spill (dry and wet conditions)

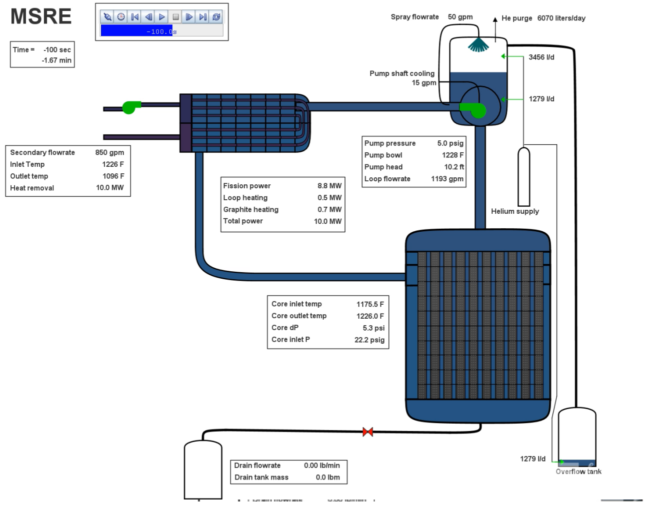
Key Insights

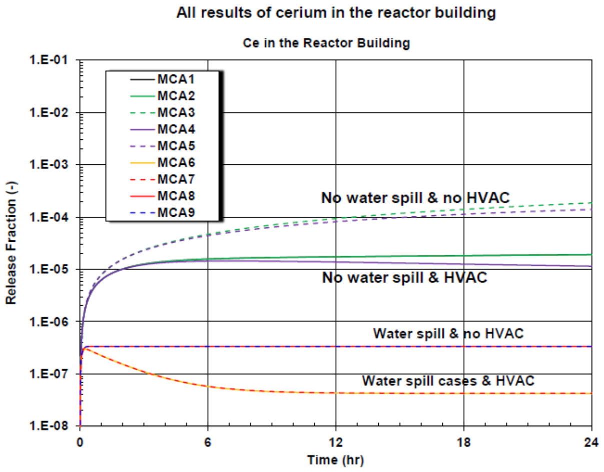
- Auxiliary filter operation increases the release of xenon to the environment while also filtering airborne aerosols
- Aerosol releases to the environment were small due to settling in the reactor cell, capture in the filter, and capture in the condensing tank in the water spill cases
- The aerosol mass in the reactor building also spanned many orders of magnitude depending on scenario assumptions











Severe Accident Analysis:
Sodium-cooled fast reactor – ABTR

<u>Reactor Characteristics</u>

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

<u>New Modeling Capabilities</u>

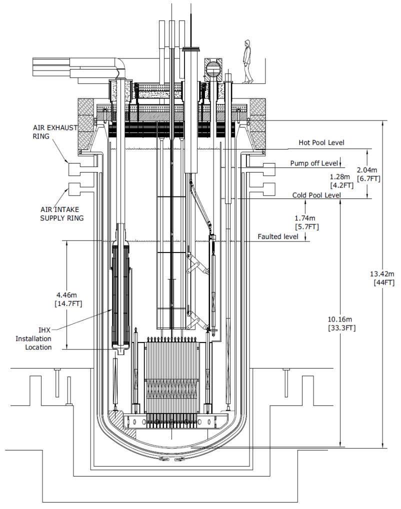
- SCALE: New capabilities in TRITON for generating nodal data for cartesian and hexagonal lattices and cells (e.g., few group homogenized cross-sections)
- MELCOR: SFR material properties, metallic fuel damage progression and radionuclide release models, sodium fire model

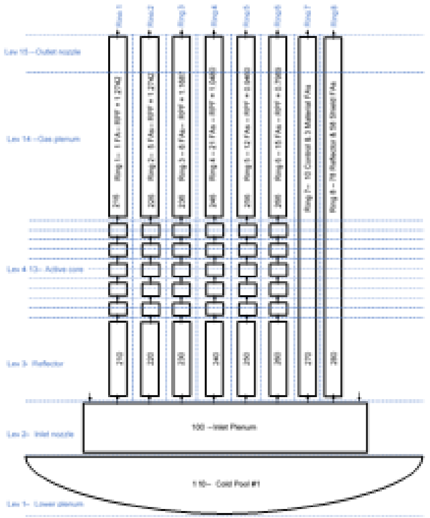
<u>Accidents Modeled</u>

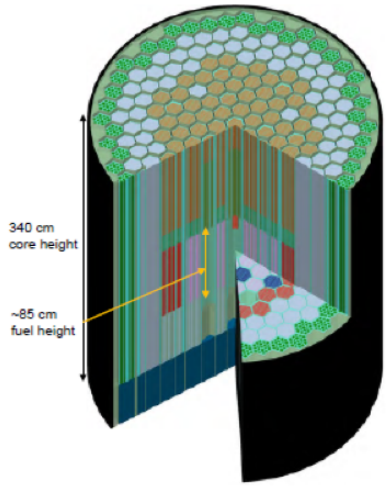
- Unprotected transient overpower, unprotected Loss-of-Flow (ULOF), and single blocked assembly

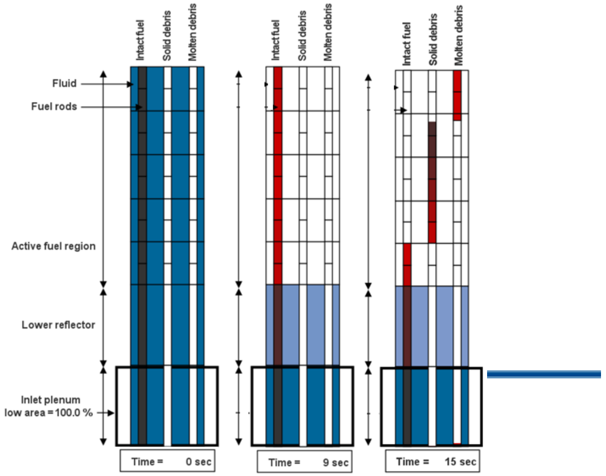
<u>Key Insights</u>

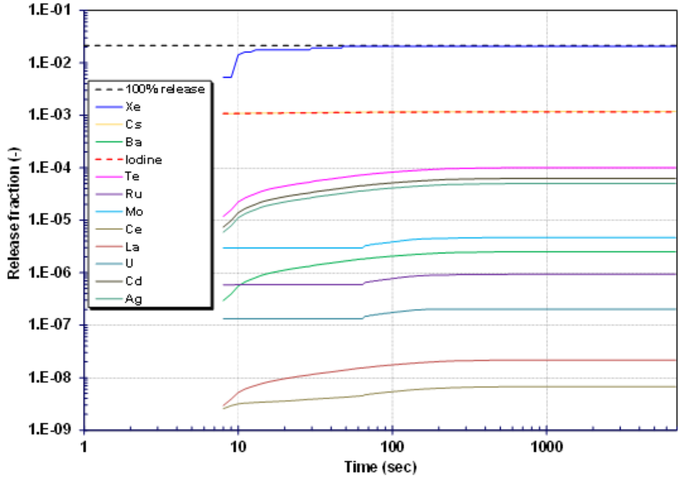
- With ULOF, core power eventually converges on the DRACS heat removal rate
- A single blocked assembly leads to rapid fuel melt







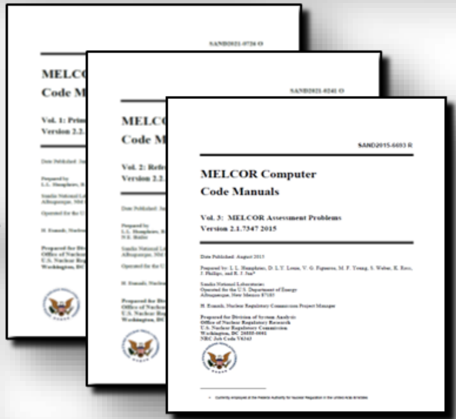


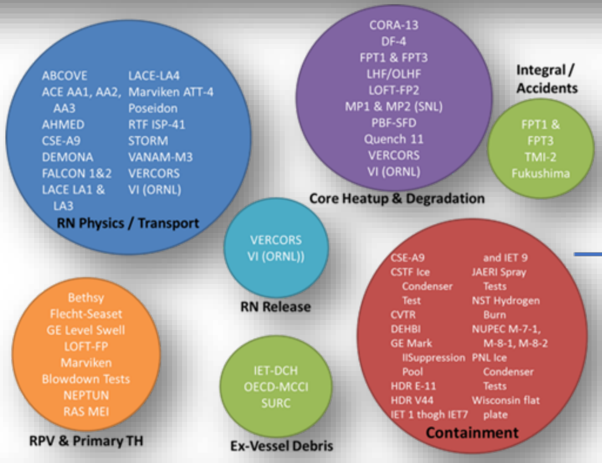


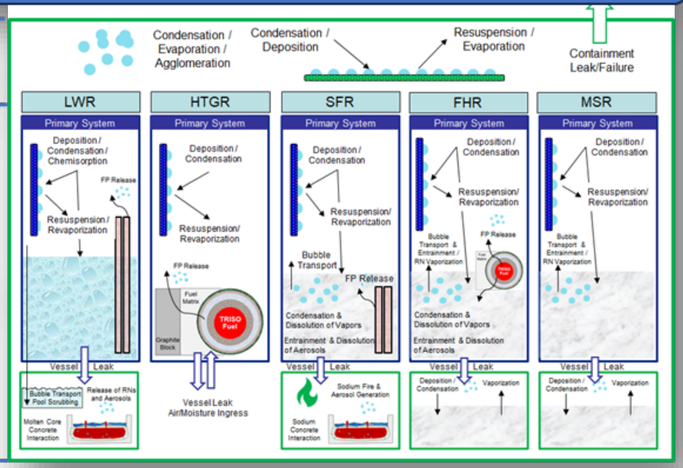
Severe Accident Analysis: MELCOR Validation & Verification Basis

Leverage existing LWR Assessment Base

Code Documentation & Assessment







Non-LWR Specific Assessment Base

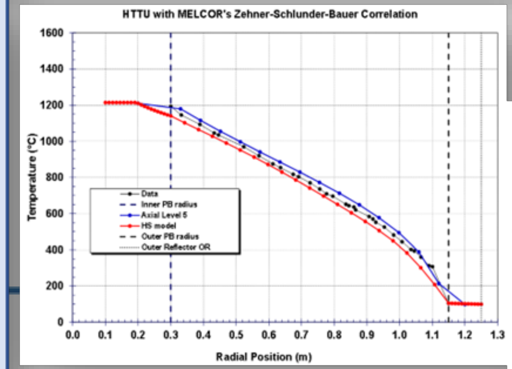
MSRE Zero-power coast-down	MSRE experiments	AB-1 AB-5 T-3	LOF,LOHS,TOP TREAT M-Series ANL-ART-38
Molten Salt (Completed)	Molten Salt (Planned)	Sodium Fires (Completed)	Sodium Reactors (Planned)

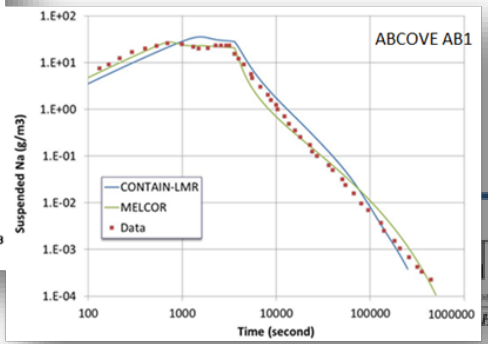
HTTU GEMINI	Air-Ingress Helical SG HT	PBMR-400 HTR-10
HTGR (Completed)	HTGR (Planned)	HTGR (In-progress)

TRISO Diffusion Release
IAEA CRP-6 Benchmark

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

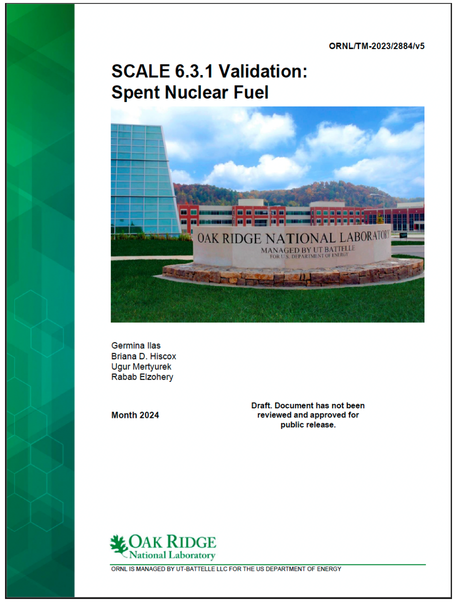
(1a): Bare kernel (1200 °C for 200 hours)
(1b): Bare kernel (1600 °C for 200 hours)
(2a): kernel+buffer+iPyC (1200 °C for 200 hours)
(2b): kernel+buffer+iPyC (1600 °C for 200 hours)
(3a): Intact (1600 °C for 200 hours)
(3b): Intact (1800 °C for 200 hours)

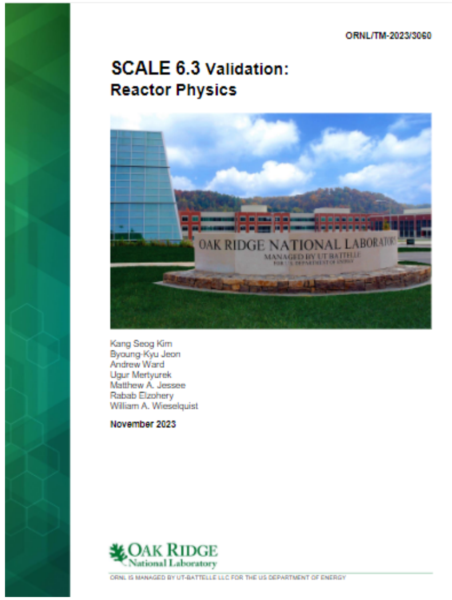


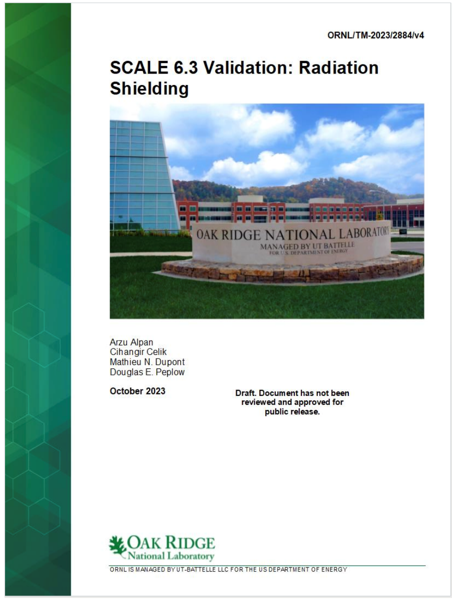


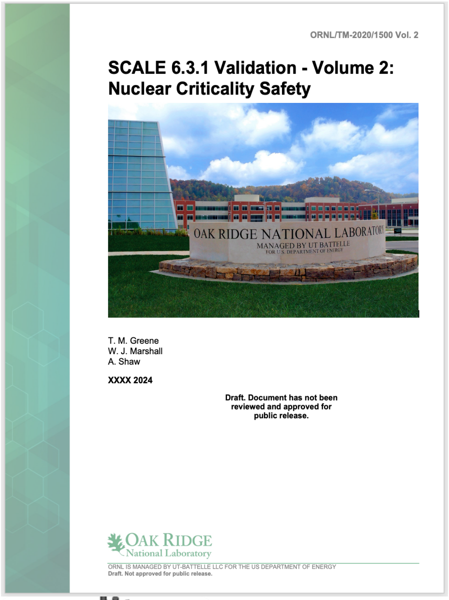
Severe Accident Analysis:
SCALE Benchmarking & Validation Activities

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









HTGRs

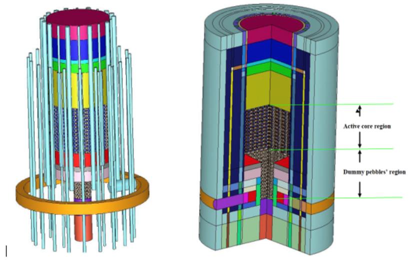


Figure 5.1. Illustration of HTR-10 benchmark model details.¶ (Channels in reflector regions [left], full reactor model [right]; images not to scale) [35].

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

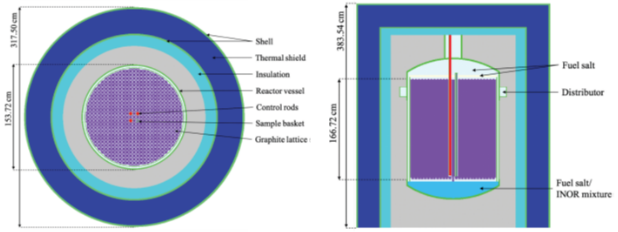


Figure 5.5. Cross-sectional illustrations of MSRE benchmark models [60].

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$_{benchmark}$).

SFRs

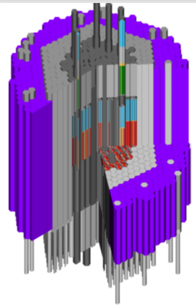


Figure 5.7. EBR-II SCALE model [62].

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$_{benchmark}$).

Severe Accident Analysis: Summary and Next Steps

1. Modeling gaps addressed through source code changes, phenomenological model development, and new analysis workflows in SCALE and MELCOR

2. SCALE & MELCOR models leveraged for supporting NRR's review of the Hermes Construction Permit Applications

3. Additional Code Enhancements & Capabilities In-Progress
   - Integration of SCALE/ORIGEN module into MELCOR for higher fidelity MSR transient analyses
   - Capability to model multiple working fluids
   - Functionality for horizontal heat pipe reactors
   - Refinement of specialized models (e.g., fluid freezing and cascading heat pipe failures)
   - Fission product chemistry refinement
   - Spatial dependence of reactivity feedback in SFRs

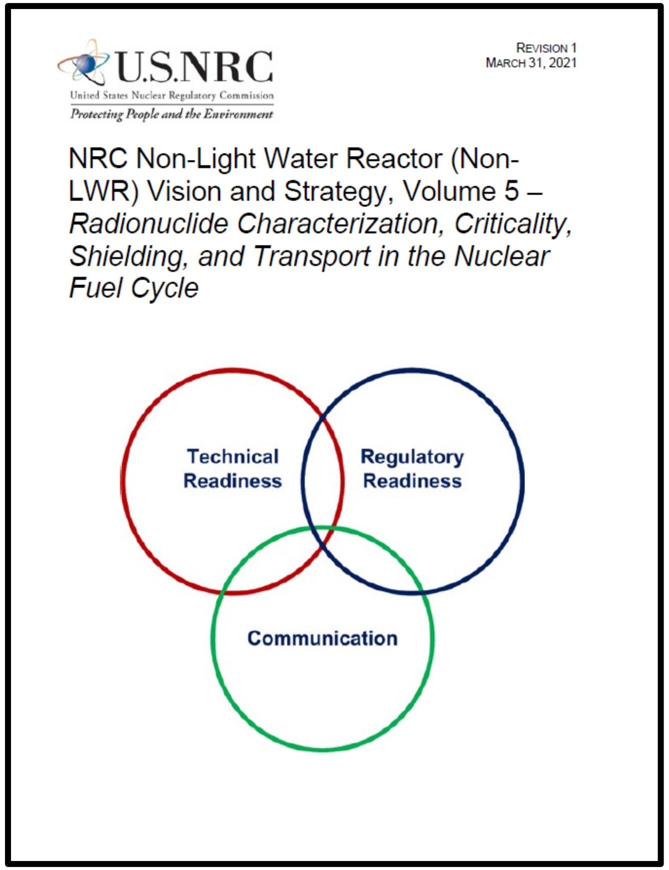
4. Data Needs

   - Validation – Criticality and depletion benchmarks that are representative of fuel designs and conditions, diffusivity of fission products, heat and mass transfer in diverse working fluids, etc.

SCALE & MELCOR code improvements and demonstration workshops have shown NRC is ready to support licensing reviews.

Volume 5

SCALE/MELCOR non-LWR Fuel Cycle Demonstration Project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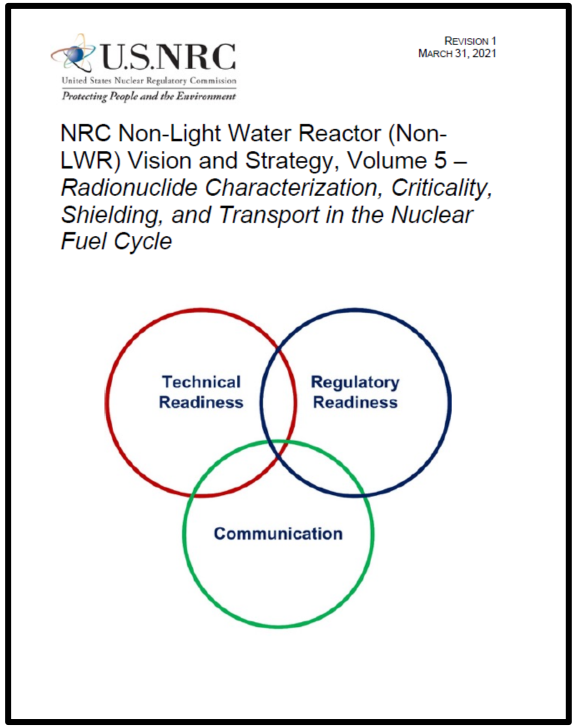
REVISION 1
MARCH 31, 2021

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

ML21088A047

Fuel Cycle Analysis: Objectives



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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*

ML21088A047

- Identify differences in potential non-LWR fuel cycles compared to LWR fuel cycle
- Identify capability gaps, in NRC's simulation capabilities (SCALE & MELCOR)
- Address any capability gaps through code development activities
- Assess, demonstrate, document through publicly available deliverables

Assess changes in the non-LWR fuel cycle & evaluate NRC's simulation capabilities for performing independent safety analyses

Fuel Cycle Analysis: Approach

- Based on publicly available information, develop models for stages of representative fuel cycles
  - Leverage the reference plants & reactor core designs from Volume 3

- Identify and select key accidents to model within SCALE & MELCOR, exercising key phenomena & models

- Develop and simulate representative SCALE & MELCOR models and evaluate
  - Identify areas where data gaps, high importance inputs, and areas to improve in our codes exist
  - SCALE – criticality, radionuclide inventory generation, decay heat, and shielding
  - MELCOR – radiological & non-radiological material & energy transport

NRC's computational capabilities will be demonstrated through public workshops and technical reports.

Fuel Cycle Analysis: Nuclear Fuel Cycle & Facility Accident Analysis

Types of Fuel Cycle Safety Analyses within Volume 5

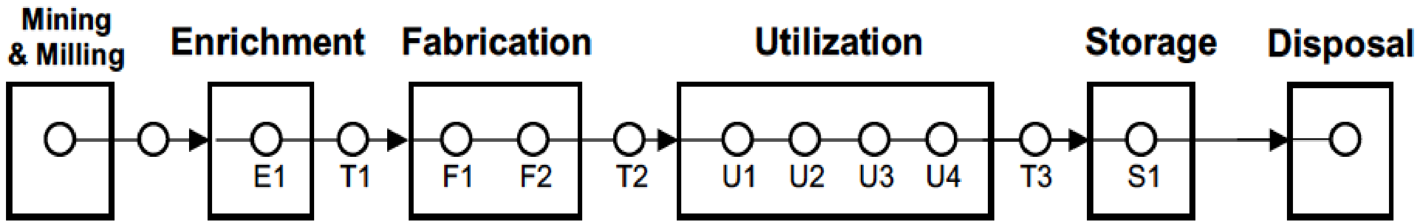
- Criticality Safety
- Radionuclide inventory & Decay heat generation
- Radiation Shielding & Dose
- Radiological material & energy release / transport
- Non-radiological material & energy release / transport

Inadvertent nuclear criticality events

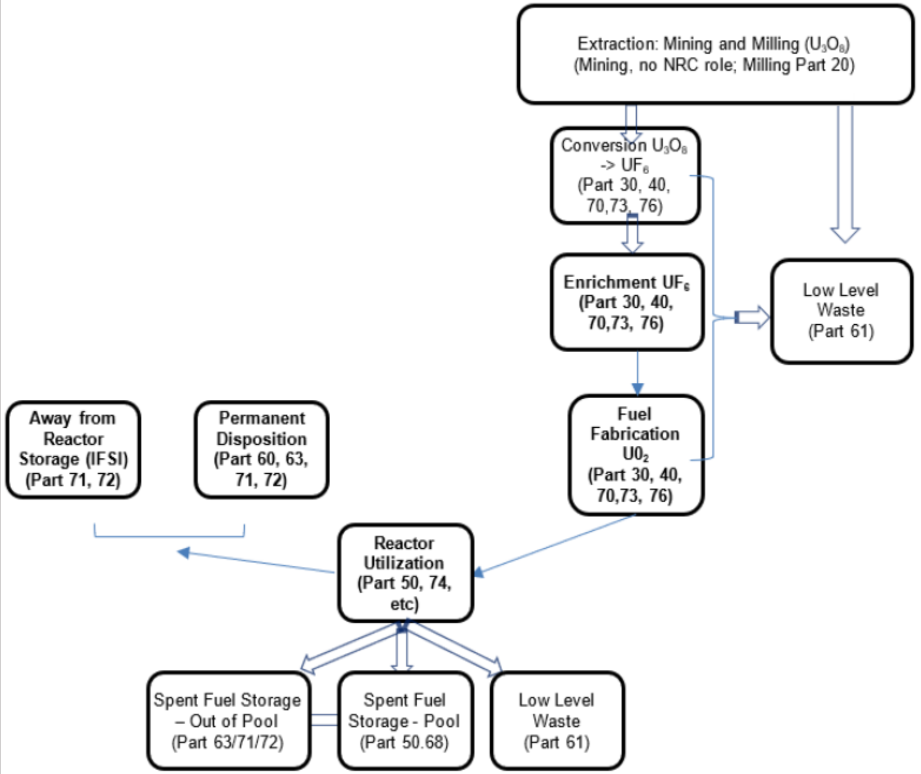
- Solution systems
- Powder systems
- Large storage arrays

NUREG/CR-6410 provides insights and methodology for performing fuel cycle safety analyses. Other references used include NUREG-1520, NUREG-2215, NUREG-2216.

Fuel Cycle Analysis: LWR Nuclear Fuel Cycle



E1 – UF_6 enrichment
T1 – transportation of UF_6 to fuel fabrication facility
F1 – fabrication of UO_2 fuel pellets
F2 – fabrication of LWR fuel assemblies
T2 – transportation of fresh fuel assemblies to the plant
U1 – fresh fuel staging and 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

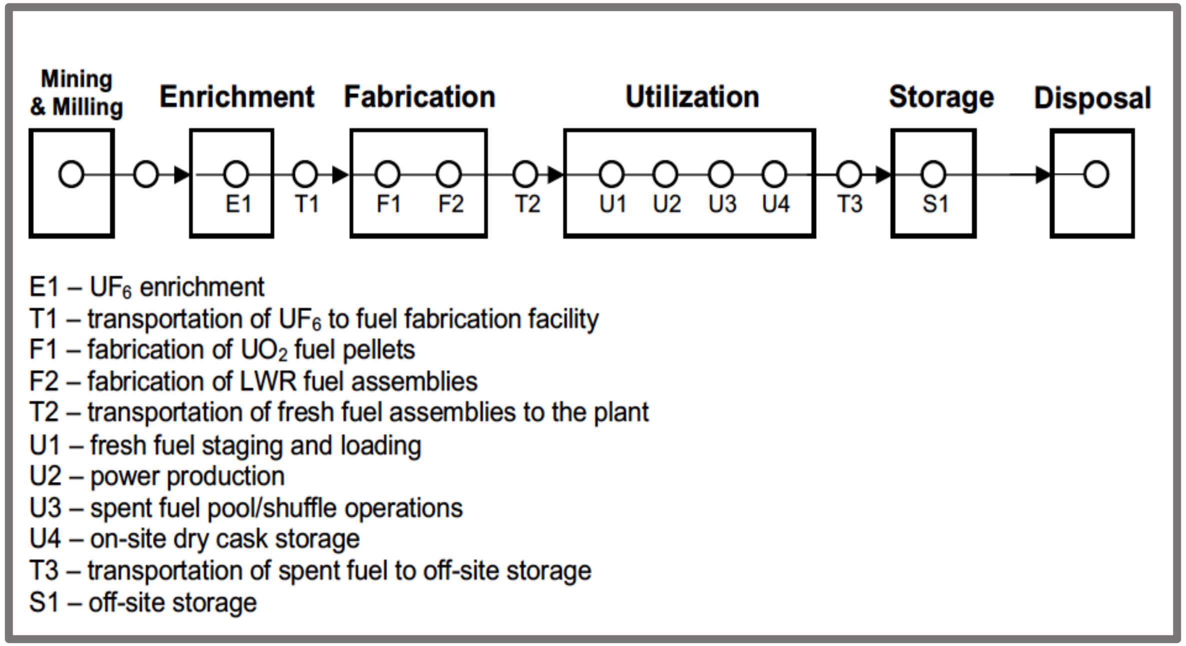


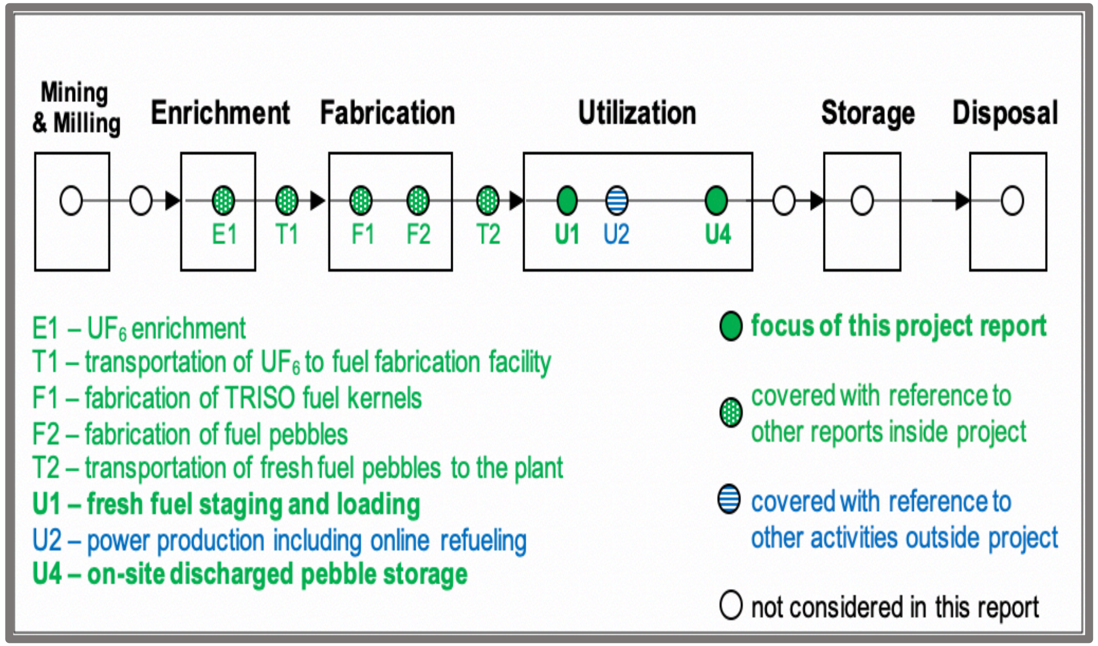
LWR open fuel cycle used as the starting point for developing each non-LWR fuel cycle.

Fuel Cycle Analysis: Non-LWR Characteristics

	Enrich (%)	Fuel Form	Approx. BU (GWd/MTU)	Fuel Residence Time	Fuel Processing	Storage	Transportation
LWRs Baseline	< 5	Uranium Oxide	62	3-4 cycles (18 - 24 months per cycle)	No	Fresh / SNF storage on site or off - site	Fresh UF6 → 30B cylinders Fresh fuel → various packages Spent fuel → various packages and dry storage systems
HPR	< 20	Oxide Metal	Up to 10	Up to 7 years	No	TBD	TBD
SFR	< 20	Metal	Up to 300	TBD	No	TBD	TBD
HTGR	< 20	TRISO pebbles TRISO compacts	100 – 200	TBD	No	TBD	TBD
FHR	< 20	TRISO pebbles TRISO compacts	100 – 200	TBD	No	TBD	TBD
MSR	< 20	Liquid	TBD	2 – 3 years	Yes	TBD	TBD

Fuel Cycle Analysis:
Non-LWR Nuclear Fuel Cycle

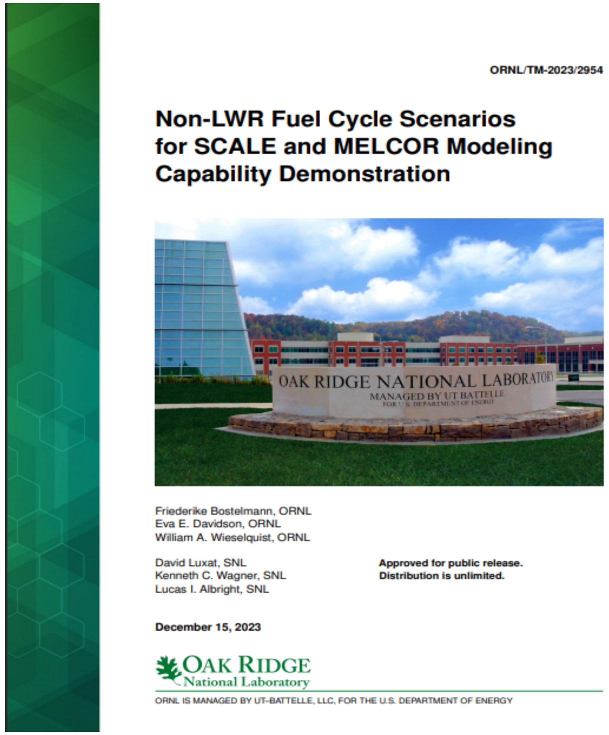




Fuel Cycle Stages Not Considered in Volume 5's Demonstration Project

Mining & Milling – No major changes envisioned from current methods.
Power Production – Executed under the Volume 3 umbrella.
Off-site Spent Fuel Storage & Transport – High degree of uncertainty for implementation.
Spent Fuel Final Disposal – High degree of uncertainty for implementation.

Fuel Cycle Analysis: Representative Fuel Cycle Designs



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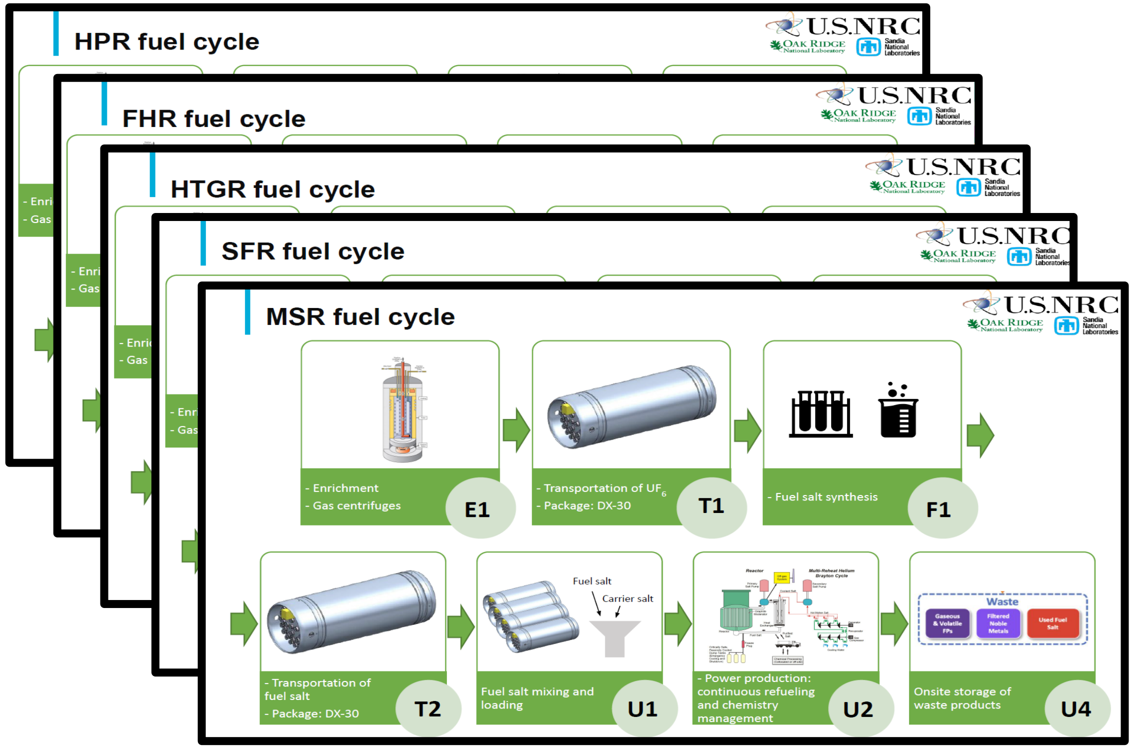


Developed five representative fuel cycle designs leveraging the Volume 3 reactor designs & identified potential accidents for the various stages of the fuel cycle.

Fuel Cycle Analysis: Types of Accidents Analyzed

Various Types of Fuel Facility Accidents

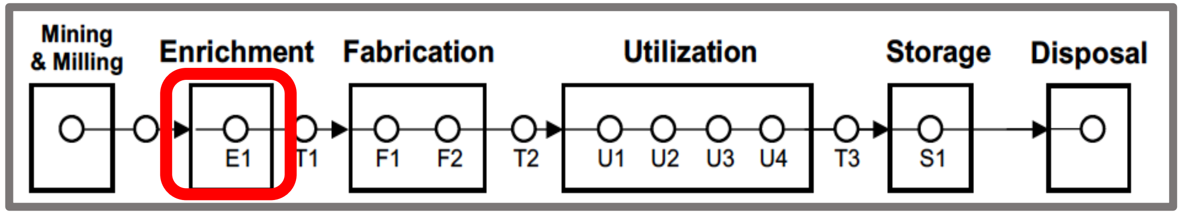
- Criticality Safety
- Radionuclide inventory & Decay heat generation
- Radiation Shielding & Dose
- Radiological material & energy release / transport
- Non-radiological material & energy release / transport

Water ingress during transportation of UF_6 in DN30-X

Dropped spent SFR fuel handling within containment during a seismic event

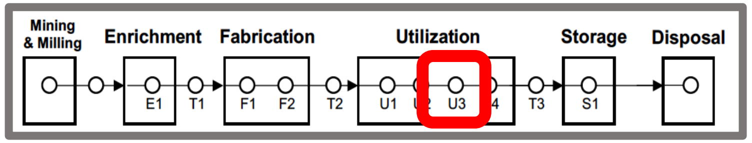
UF_6 cylinder is overfilled and heated, resulting in a rupture

Fuel Cycle Analysis: Highlights - UF_6 Enrichment



Hazardous Material Identified	Potential Accidents
Inventory of hazardous chemicals identified (NH_3, F_2, HF, KOH, UF_6) UF_6 identified as the only source of dispersible radiological material in this fuel cycle stage.	Radiological Release ➢ UF_6 cylinder rupture (overfill/heated, damage/drop) Criticality Safety ➢ UF_6 criticality up to HALEU enrichment Non-radiological ➢ HF, NH_3, F_2 release (seismic / pipe rupture)

Fuel Cycle Analysis:
SFR Fuel Handling Accident - Dose



During refueling operations, the refueling machine is used to perform fuel handling operations, such as moving spent fuel assembly in and out of the reactor core. A seismic event occurs causing the refueling machine to fail and drop a spent fuel assembly within the containment building.

- SCALE is used to determine the spent fuel nuclear inventory and perform the radiation dose estimates throughout the containment building. The radiation dose rate (radiative source term) is based upon an intact fuel assembly at various cooling periods.
- Modeling Assumptions
  - Spent fuel assembly is intact.
  - Containment building consists of a 1.2 cm thick steel liner, with reinforced concrete (1 m). Rebar-to-concrete mass ratio is 0.106.

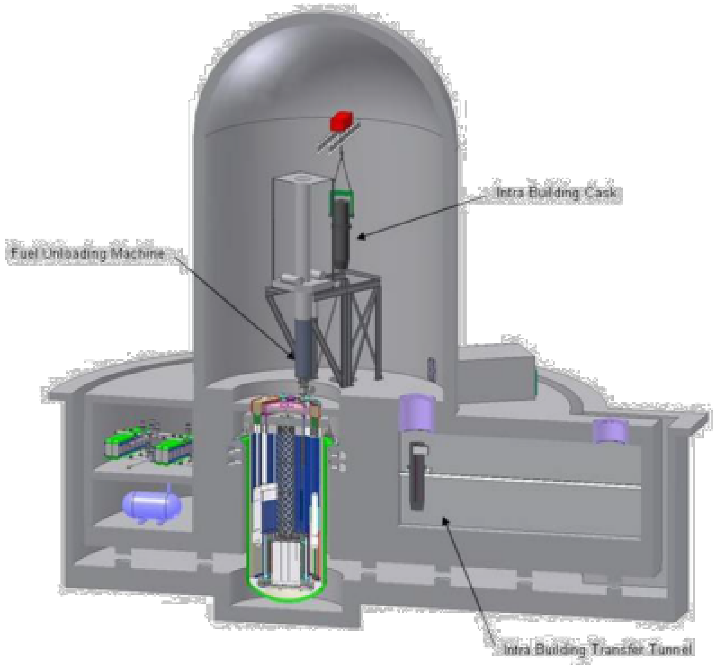
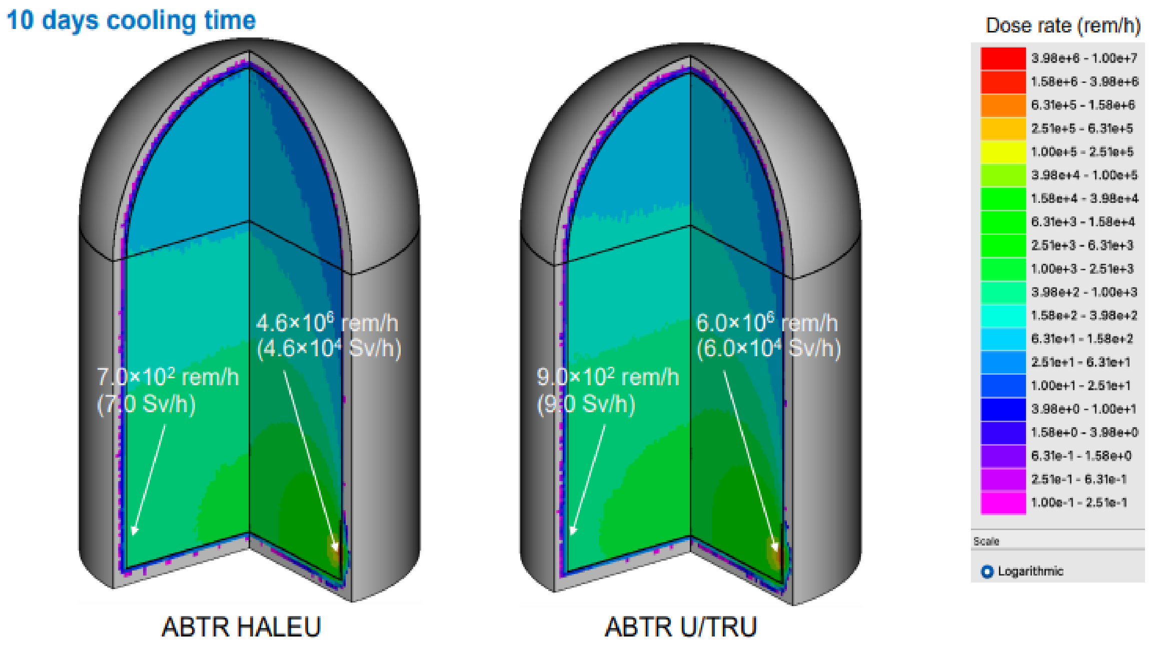
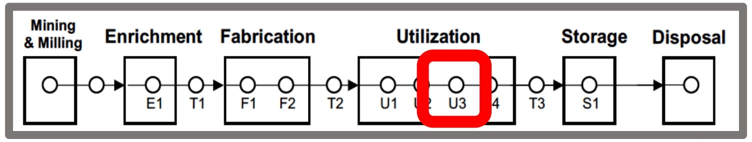


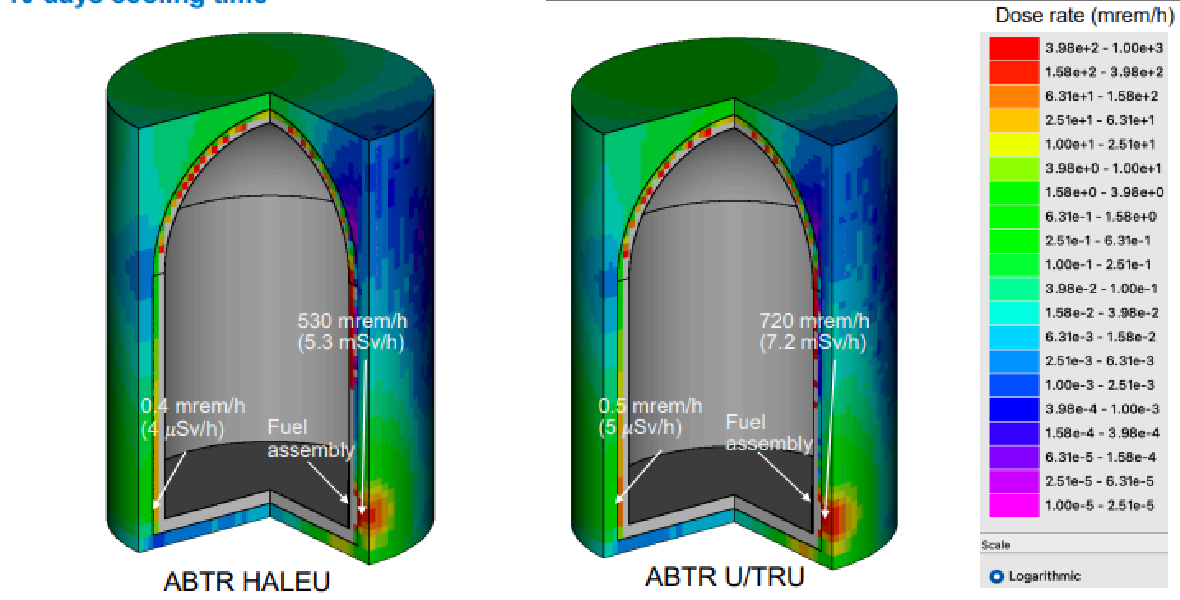
Figure II.7-1 Schematic of Fuel Handling System

ABTR reactor building

Fuel Cycle Analysis: SFR Fuel Handling Accident - Dose





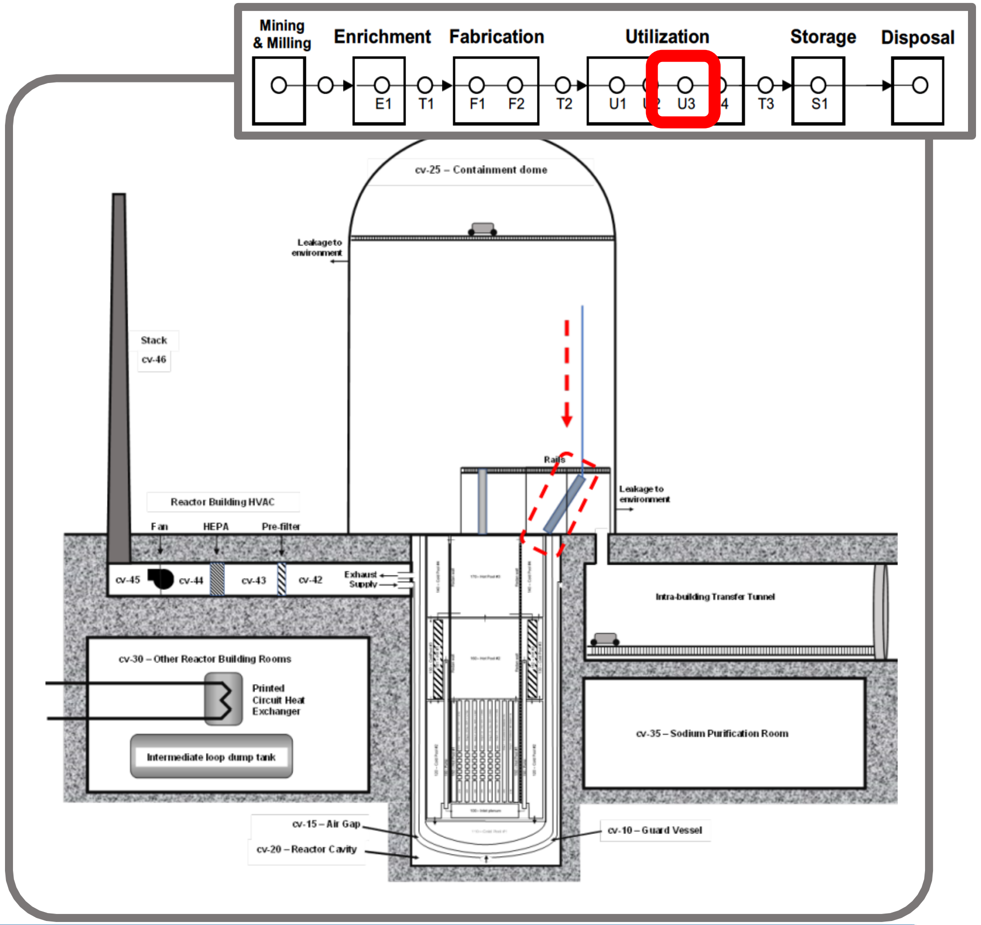


- Two cases analyzed; fuel cooled for 10 days & 7 reactor cycles.
  - 7 reactor cycles is the length of time a fuel assembly (FA) remains in the in-vessel storage.
- Neutron and gamma source terms determined for both ABTR HALEU & ABTR U/TRU fuel types.

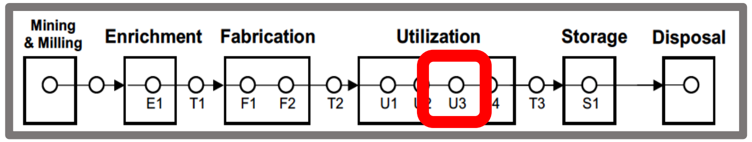
Fuel Cycle Analysis: SFR Fuel Handling Accident –Material Transport

During refueling operations, the refueling machine is used to perform fuel handling operations, such as moving spent fuel assembly in and out of the reactor core.
A seismic event occurs causing the refueling machine to fail and drop a spent fuel assembly loaded within a SNF cask within the containment building.

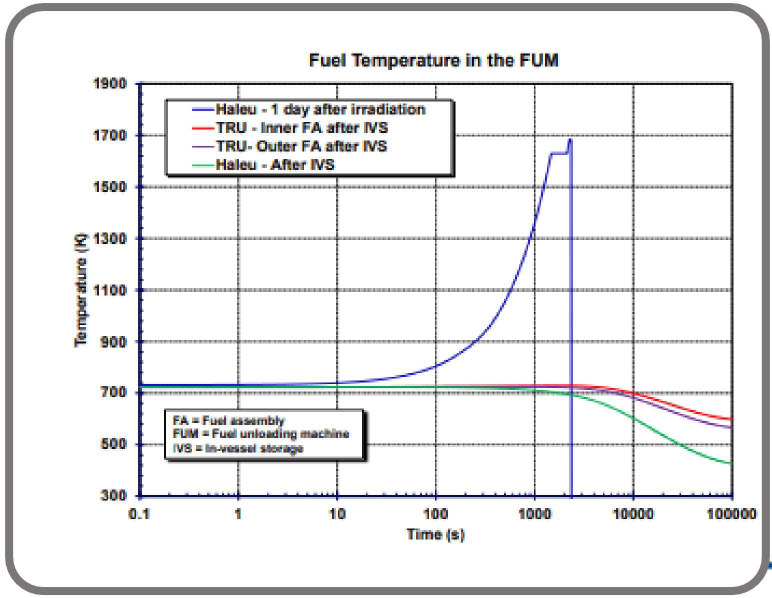
- MELCOR is used to model the fuel damage and radiological transport throughout the containment building. SCALE is used to provide the radionuclides for the HALEU spent fuel after in-vessel storage (7 cycles).
- Modeling Assumptions
  - No residual sodium in the cask.
  - All active cooling systems have failed.

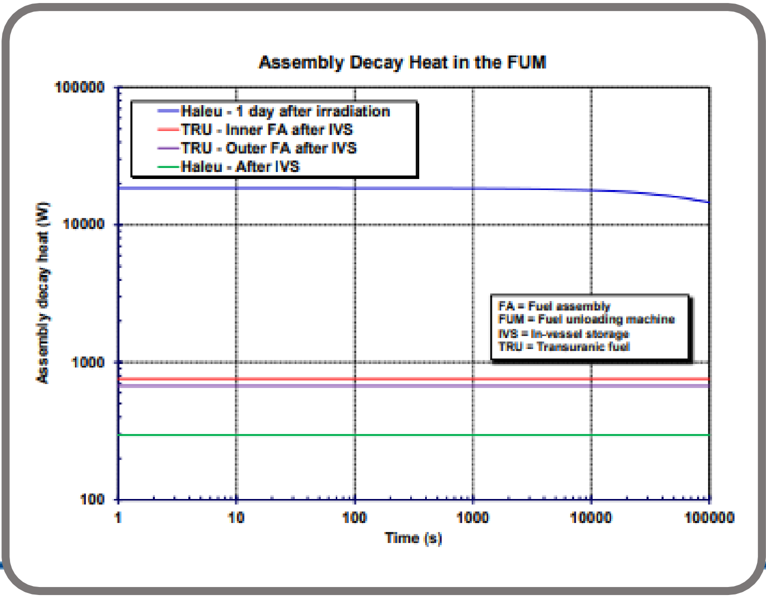


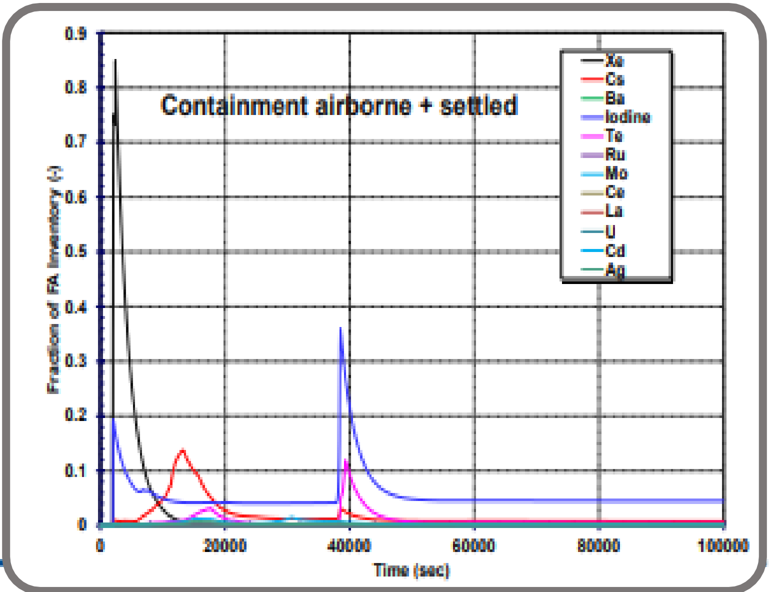
Fuel Cycle Analysis:
SFR Fuel Handling Accident –Material Transport



- During removal from the reactor, FA are blown with argon gas to remove residual sodium.
- FAs with normal in-vessel storage cooling times remain intact within the failed fuel handling machine
- Accidental removal of a recently discharged FA would lead to fuel failures after 40







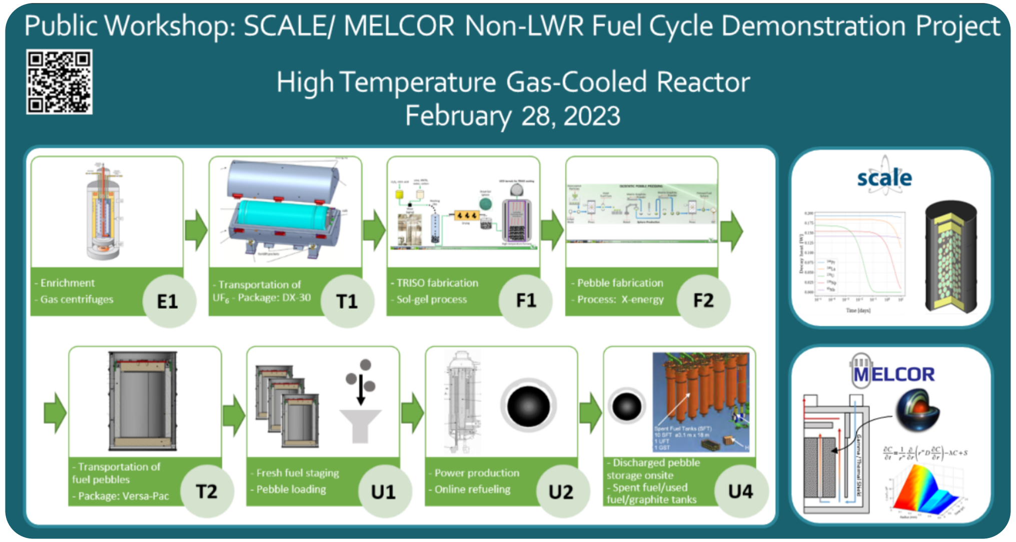
Fuel Cycle Analysis: Public Workshops & Webpage

SCALE/MELCOR non-LWR fuel cycle demonstration project	

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

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

• Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration ◦ Report	December 15, 2023



Next planned workshop will be held Spring 2025 that discusses a non-LWR horizontal heat pipe reactor (microreactor). Workshop will cover in-reactor accident progression & fuel cycle analyses.

Fuel Cycle Analysis: Key Highlights & Conclusions

- Workshops and analyses have revealed some information gaps, for example:
  - No commercially-sized transportation packages for moving fresh pebbles.
  - Lack of public information for onsite fresh & spent fuel storage (pebbles, SFR fuel, etc.).

It is not envisioned this will challenge SCALE/MELCOR since no new models are required.

- The need for validation data (criticality safety benchmarking) has been identified, especially for TRISO based systems.
  - New collaboration between DOE and NRC for the Development of Criticality Safety Benchmarking Data for HALEU Fuel Cycle and Transportation (DNCSH)
  - Goal is to produce high-quality publicly available benchmarking experiments, nuclear data, and evaluations applicable to a wide range of HALEU systems.

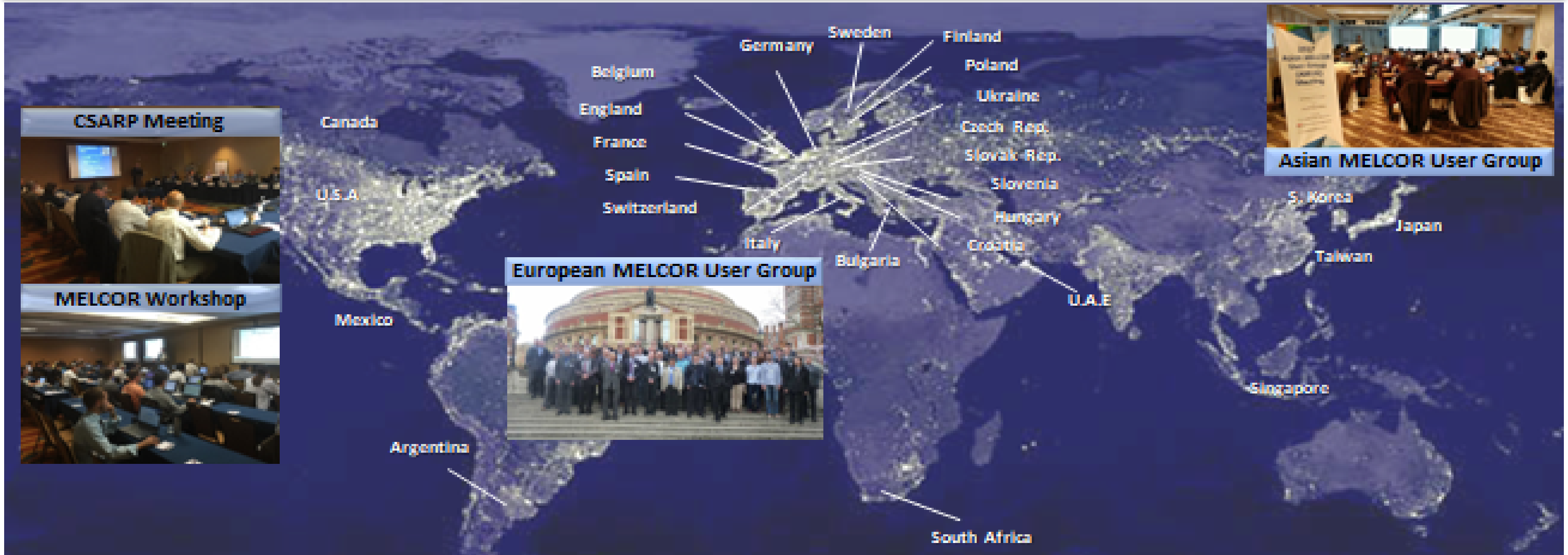
SCALE & MELCOR demonstration workshops have shown NRC is ready to support fuel cycle analyses

Fuel Cycle Analysis: Next Steps

- Code development activities ongoing
  - MELCOR/ORIGEN Integration for MSR analyses
  - Capability to model multiple working fluids in the same MELCOR plant model
  - Addition of limited unstructured mesh capability to allow analysis of complex, arbitrary geometries of fissile material (e.g., fractured / damaged TRISO pebbles) in SCALE.
  - Improved modeling capabilities in SCALE to control-blades within pebble bed systems.
- Maintain awareness of industry priorities
- Training and knowledge management

CSARP International Cooperation

Cooperative Severe Accident Research Program (CSARP) – Summer/U.S.A (June 3-5, 2024)
MELCOR Code Assessment Program (MCAP) – Summer/U.S.A (June 6-5, 2024)
European MELCOR User Group (EMUG) Meeting – Spring/Europe
Asian MELCOR User Group (AMUG) Meeting – Fall/Asia



NUREG/BR-0524, "Cooperative Severe Accident Research Program (CSARP)"

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Questions?