SCALE & MELCOR non-LWR Fuel Cycle Demonstration Project

Sodium Fast Reactors

NRC's Volume 5 – Public Workshop #2

September 20, 2023

U.S. Nuclear Regulatory Commission

Office of Nuclear Regulatory Research

Office of Nuclear Reactor Regulations

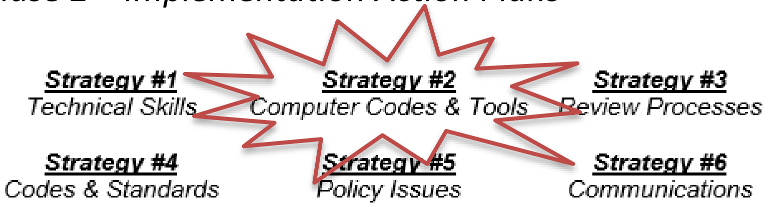
Office of Nuclear Material Safety and Safeguards

Outline

- NRC Strategy for non-LWRs Readiness
- Project Scope
- SFR Nuclear Fuel Cycle
- Overview of the Simulated Accidents
- Nuclide inventory, decay heat, and criticality calculations in SCALE
- Sodium Fast Reactor Modeling using MELCOR
- Summary & Closing Thoughts

NRC’s Strategy for Preparing for non-LWRs

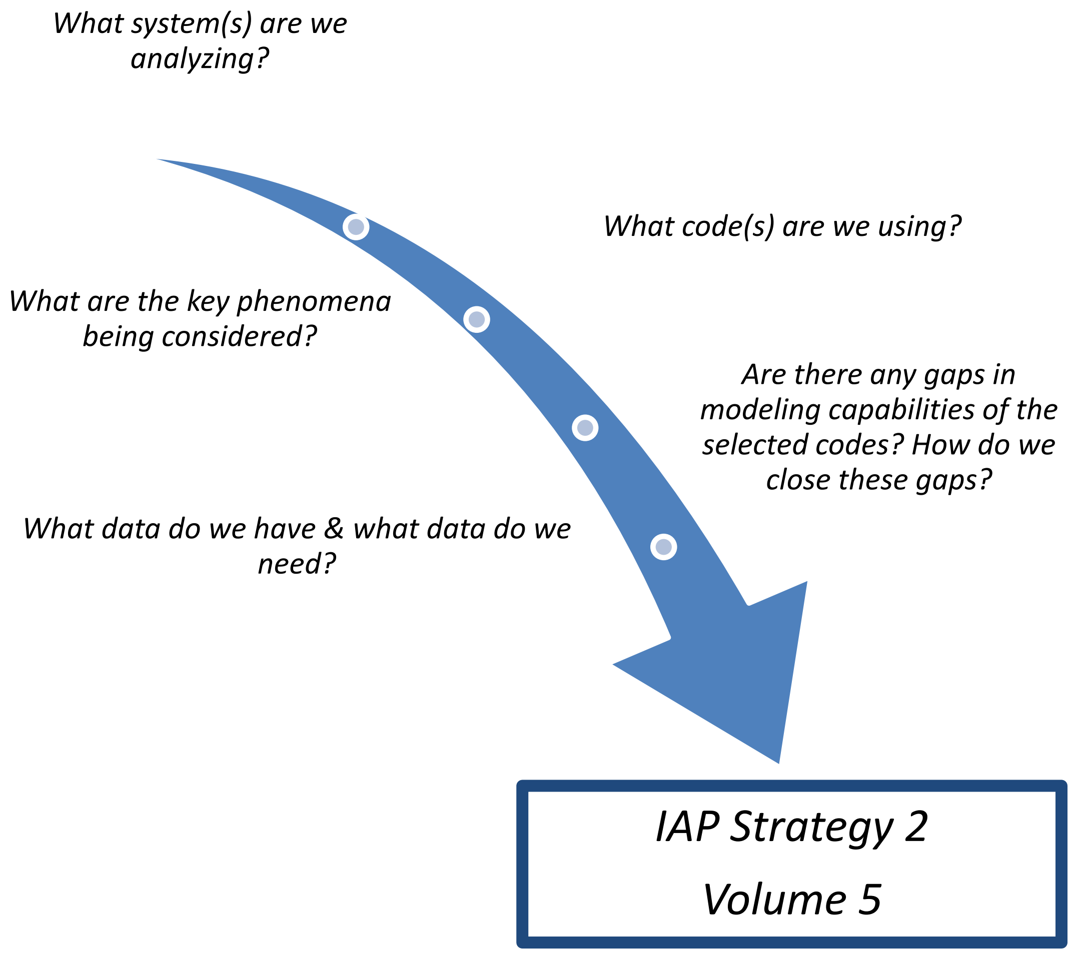
- *NRC’s Readiness Strategy for Non-LWRs*
  - *Phase 1 – Vision & Strategy*
  - *Phase 2 – Implementation Action Plans*

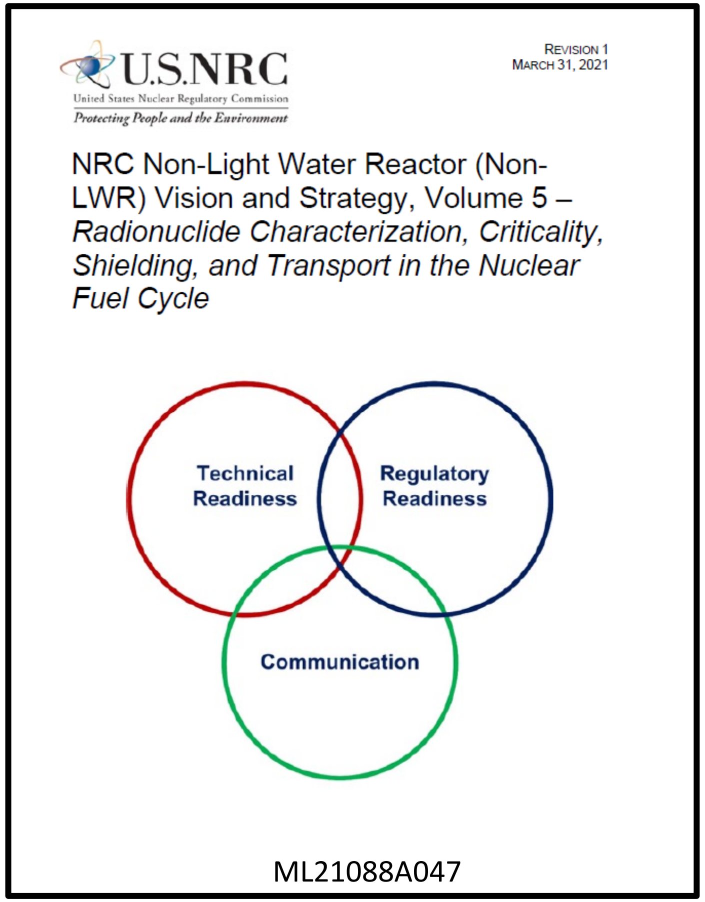


- *IAPs are planning tools that describe:*
  - *Required work, resources, and sequencing of work to achieve readiness*

- *Strategy #2 – Computer Codes and Review Tools*
  - *Identifies computer code & development activities*
  - *Identifies key phenomena*
  - *Assess available experimental data & needs*

What's in Volume 5?





U.S.NRC
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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*

Technical Readiness
Regulatory Readiness
Communication

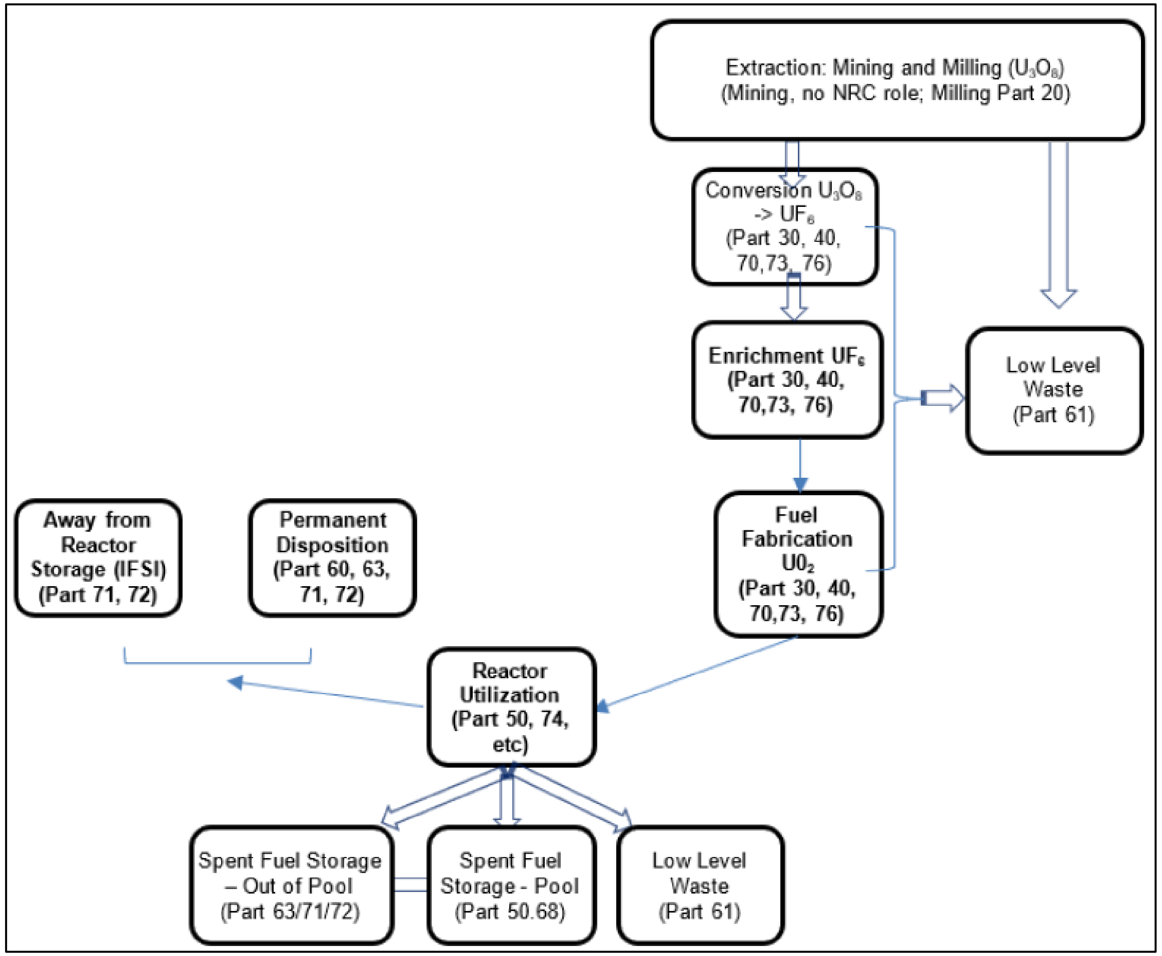
ML21088A047

LWR Nuclear Fuel Cycle

Regulations for the Nuclear Fuel Cycle

- *Protects onsite workers, public and the environment against radiological and non-radiological hazards that arise from fuel cycle operations.*
  - *Radiation hazards*
  - *Radiological hazards*
  - *Non-radiological (i.e., chemical) hazards*

- *Applicable Regulations*
  - *Uranium Recovery / Milling – 10 CFR Part 20*
  - *Uranium Conversion – 10 CFR Parts 30, 40, 70, 73 and 76*
  - *Uranium Enrichment – 10 CFR Parts 30, 40, 70, 73 and 76*
  - *Fuel Fabrication – 10 CFR Parts 30, 40, 70, 73 and 76*
  - *Reactor Utilization – 10 CFR Parts 50 & 74*
  - *Spent Fuel Pool Storage – 10 CFR Parts 50.68*
  - *Spent Fuel Storage (Dry) – 10 CFR Parts 63, 71, and 72*



Project Scope - Non-LWR Fuel Cycle

- *Stages in scope for Volume 5*

1. Enrichment — *UF_6 enrichment*
2. UF_6 Transportation
3. Fuel Fabrication
4. Fresh Fuel Transportation
5. Fuel Utilization (including on-site spent fuel storage)

- *Stages out of scope for Volume 5*

Uranium Mining & Milling	• *Not envisioned to change from current methods.*
Power Production	• *Successfully completed and leveraged from the* ***Volume 3 – Source Term & Consequence work***
Spent Fuel Off-site Storage & Transportation	• *Large amount of uncertainties for non-LWR concepts & lack of information*
Spent Fuel Final Disposal	• *Large amount of uncertainties for non-LWR concepts & lack of information*

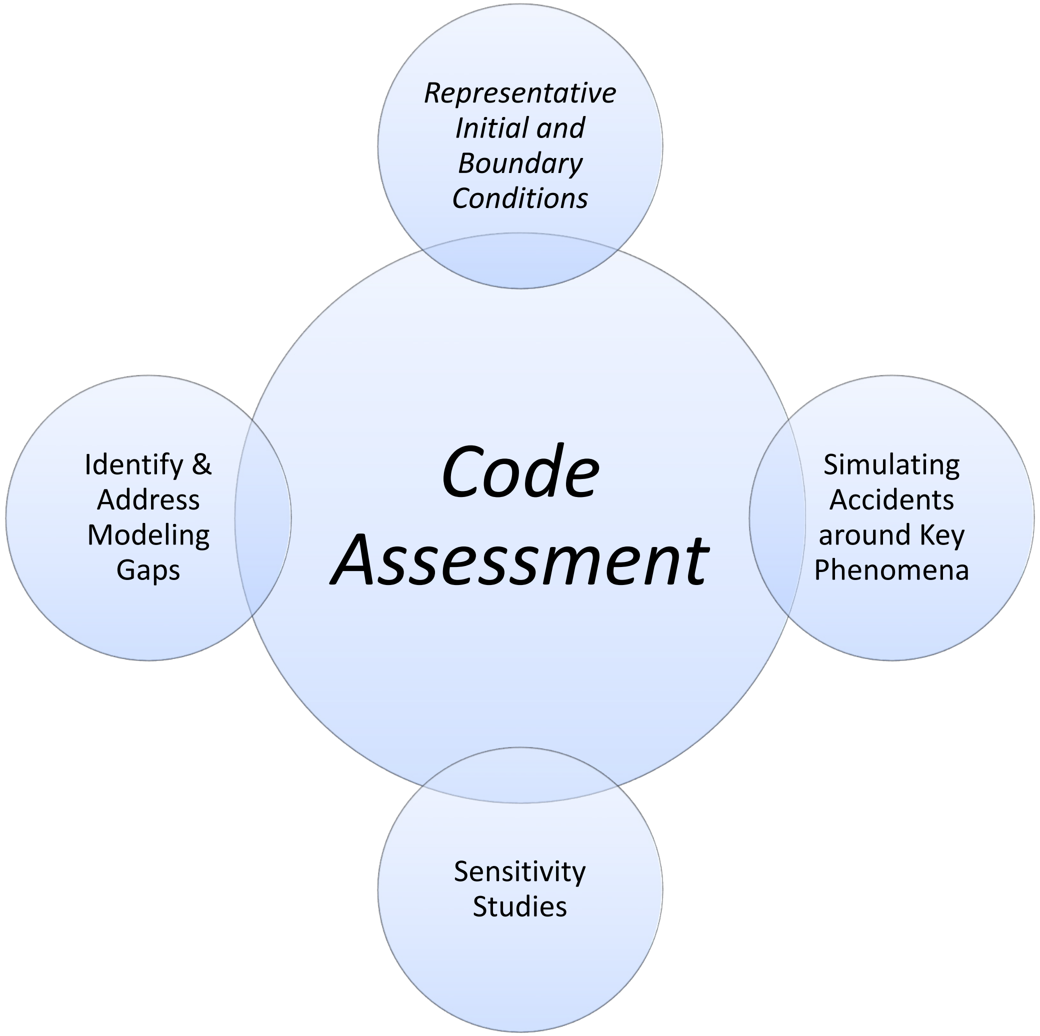
Codes Supporting non-LWR Nuclear Fuel Cycle Licensing

- NRC's comprehensive neutronics package
  - Nuclear data & cross-section processing
  - Decay heat analyses
  - Criticality safety
  - Radiation shielding
  - Radionuclide inventory & depletion generation
  - Reactor core physics
  - Sensitivity and uncertainty analyses

- NRC's comprehensive accident progression and source term code
  - Characterizing and tracking accident progression,
  - Performing transport and deposition of radionuclides throughout a facility,
  - Performing non-radiological accident progression

Project Approach

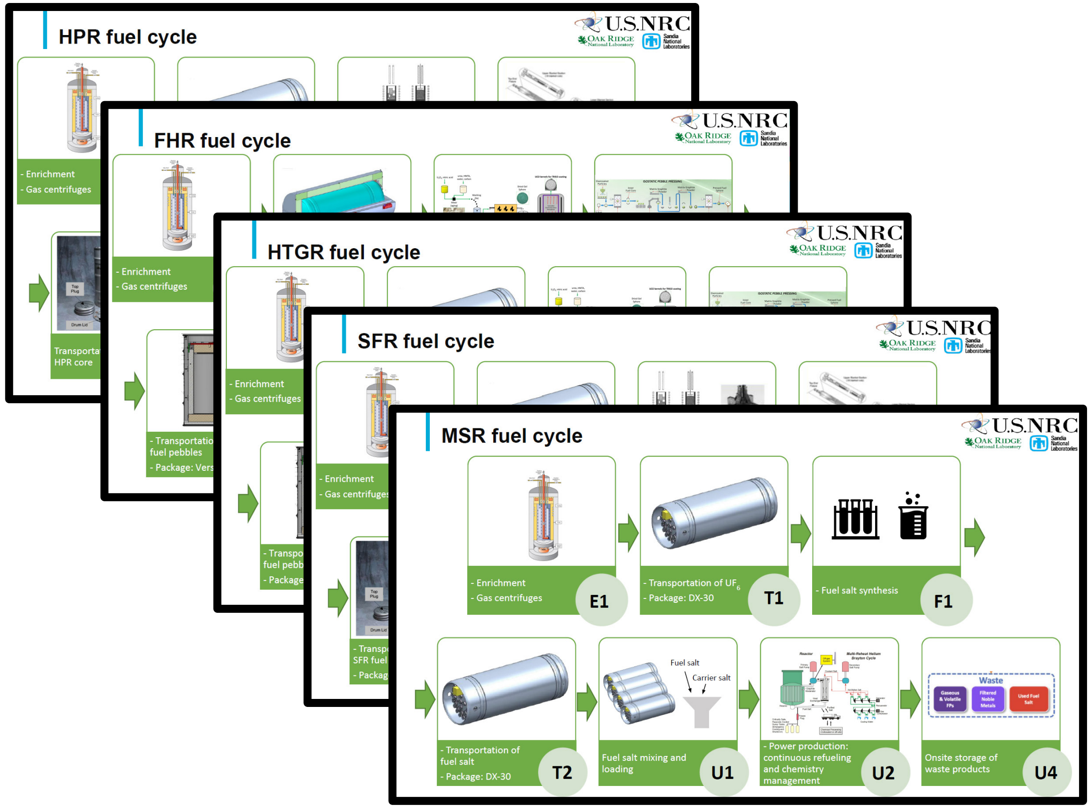
- Build representative fuel cycle designs **leveraging** the Volume 3 designs
- Identify key scenarios and accidents exercising key phenomena & models
- Build representative SCALE & MELCOR models and evaluate



Representative Fuel Cycle Designs

- Completed 5 non-LWR fuel cycle designs for –
  - *Heat Pipe Reactor (HPR)– INL Design A*
  - *High Temperature Gas Reactor (HTGR) – Pebble Bed Modular Reactor (PBMR)-400*
  - *Fluoride-Salt Cooled Hight Temperature Reactor (FHR) – University of California, Berkeley (UCB) Mark 1*
  - *Molten Salt Reactor (MSR) – Molten Salt Reactor Experiment (MSRE)*
  - *Sodium-Cooled Fast Reactor (SFR) - Advanced Burner Test Reactor (ABTR)*

- Identifies potential processes & methods, for example:
  - *What shipping package could transport HALEU-enriched UF6? What are the hazards associated?*
  - *How is spent SFR fuel moved? What are the hazards associated?*
  - *How is fissile salt manufactured for MSRs? What are the various kinds of fissile salt that may be used? What are the hazards?*



Prototypic Initial and Boundary Conditions for the SCALE & MELCOR Analyses

Overview of the SFR fuel cycle

F. Bostelmann

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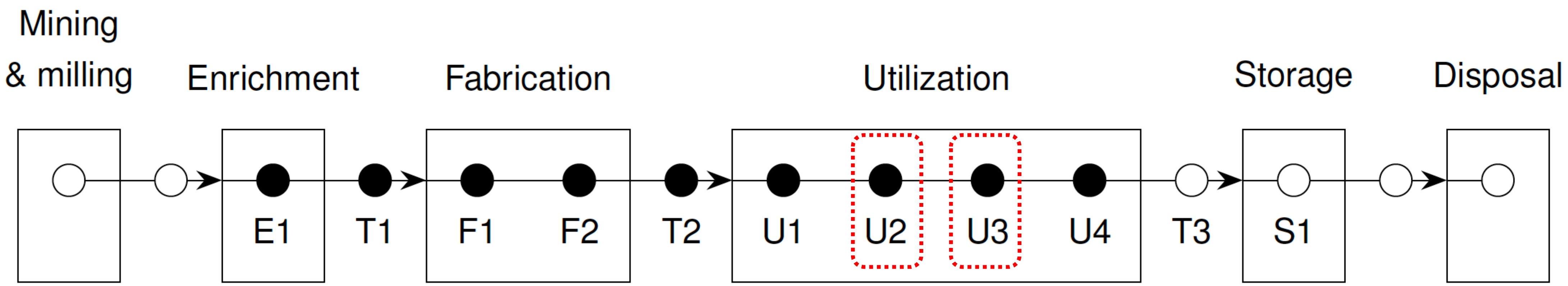
Overview

Initial project effort was to identify hazards across the SFR fuel cycle

- Determine details of the fuel cycle stage based on **publicly** available information
  - Use ABTR as basis for fuel assembly details and for SFR operation
  - Consider metallic SFR fuel
- Identify potential hazards and accident scenarios for each stage of the fuel cycle
  - Identify accidents independently of their probability for occurrence
- Select accident scenarios to demonstrate SCALE/MELCOR's capabilities

SFR Fuel Cycle with Once-Through Fuel



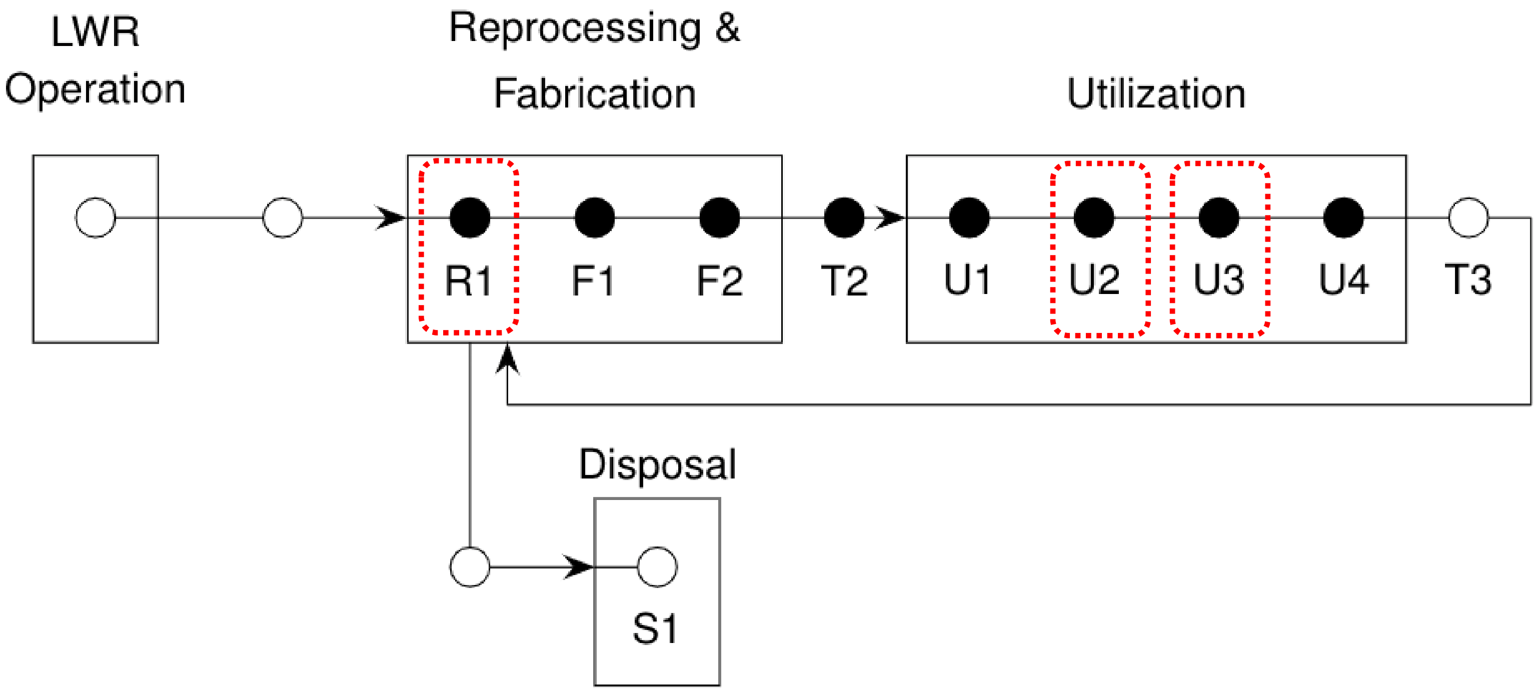
E1 – UF_6 enrichment
T1 – Transportation of UF_6 to fabrication facility
F1 – Fuel fabrication
F2 – Fuel assembly/pebble fabrication
T2 – Transportation of assemblies/pebbles/salt to plant
U1 – Fresh fuel staging/preparation/loading

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

● Considered in this work
○ Not considered in this work

Scenario for this stage studied in this workshop

SFR Fuel Cycle with Reprocessed Fuel



R1 – Reprocessing
F1 – Fuel fabrication
F2 – Fuel assembly fabrication
T2 – Transportation of assemblies/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
S1 – Off-site storage of waste products

● Considered in this work
○ Not considered in this work

Scenario for this stage studied in this workshop

E1: Enrichment

- Enrichment of UF_6 up to 19.75 wt.% ^{235}U [High Assay Low-Enriched Uranium (HALEU)]
- US facilities for uranium enrichment using gas centrifuges
  - Louisiana Energy Services (Urenco USA) in Eunice, NM
    - Currently the only active commercial process for enrichment of up to 5 wt.% ^{235}U in the US
  - Centrus Energy Corp in Piketon, OH
    - First U.S. facility licensed for HALEU production
    - DOE program, started in 05/19, revised in 03/22
      - Phase 1 (~1 year): installation of HALEU cascade, demonstration of production of 20 kg UF_6 HALEU
      - Phase 2 (1 year): production of 900 kg UF_6 HALEU
      - Phase 3 (3 year): production of 900 kg UF_6 HALEU/year

Major hazards:

- UF_6 liquid and vapor leaks from damaged pipes or cylinders
- Criticality due to unintended accumulation of enriched U

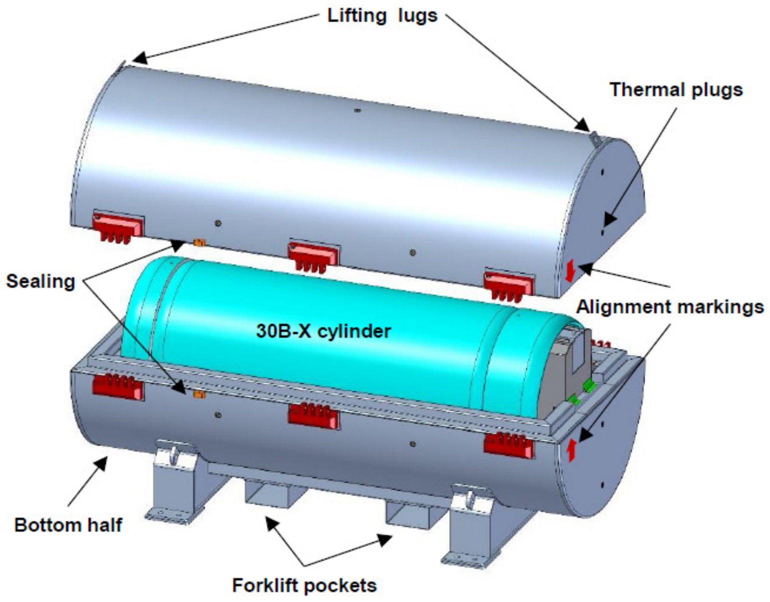
T1: Transportation of UF_6

ORANO DN30-X package for up to 20 wt% ^{235}U enrichment:

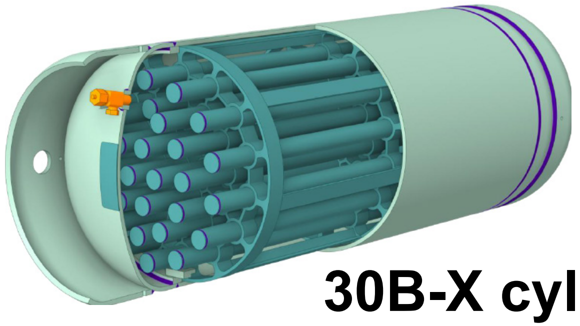
- 30B-X cylinder similar to 30B cylinder, but with criticality control system (internal absorber structure)
- Permissible mass in DN30-X:

Package design	Enrichment limit	Permissible UF_6 mass
DN30-10	10 wt.% ^{235}U	1460 kg
DN30-20	20 wt.% ^{235}U	1271 kg

- DN30-X protective structural packaging (PSP) unchanged to DN30: outer PSP acts as a shock absorber during drop tests and as thermal protection in fire tests



DN30-X package



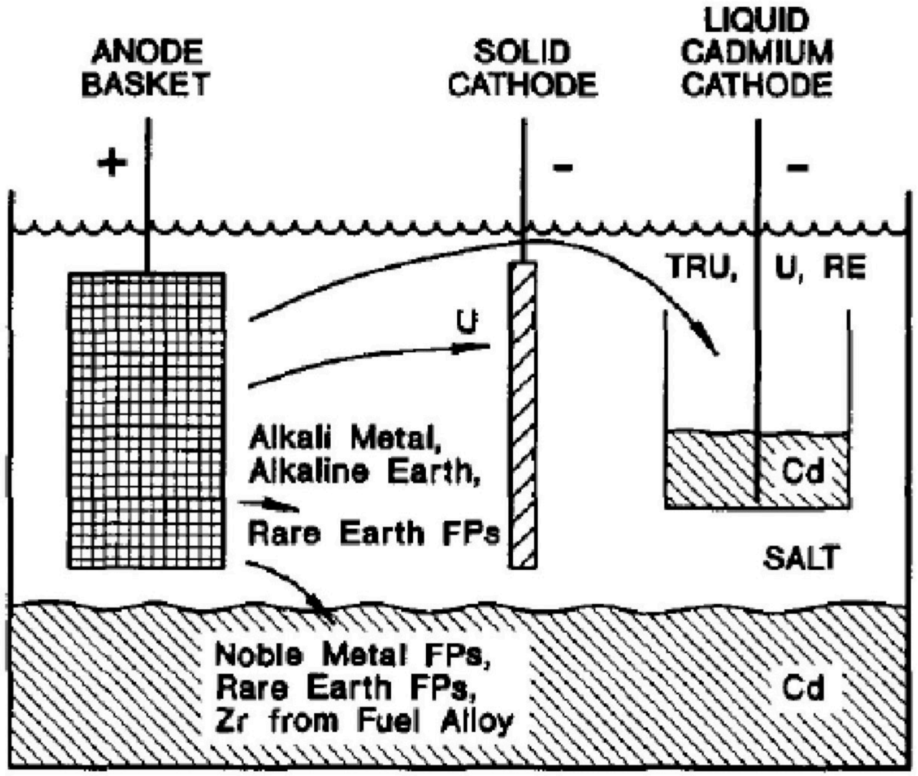
30B-X cylinder

Major hazards:
- Criticality due to water accidents and container drop
- Release of UF_6 due to container rupture

Ref.: ORANO Safety Analysis Report for the DN30-X Package
https://www.nrc.gov/docs/ML2232/ML22327A183.pdf
Certificate of Compliance, Certificate number 9388
https://rampac.energy.gov/docs/default-source/certificates/1019388.pdf

R1: Reprocessing of Spent Nuclear Fuel

- Reprocessing currently not pursued in the US, but only considered here to demonstrate code capabilities
- Electrometallurgical treatment technology was originally proposed by ANL and already performed for EBR-II fuel
- Electrometallurgical processing:
    - Complete set of operations to capture actinide elements from spent fuel and recycle them as fuel materials
- Process:
    - Steel vessel with cadmium layer and electrolyte salt at 500°C
    - Chopped fuel is loaded into the anode basket
    - Actinides transport via electric current
    - Cathode deposits (U/Pu) are consolidated by melting and ready for to be used in fuel slug fabrication



Schematic of the electrometallurgical treatment used for metallic fuel from the EBR-II

Major hazards:

- Criticality from misfeeding or mishandling of fuel
- Release of radiological materials

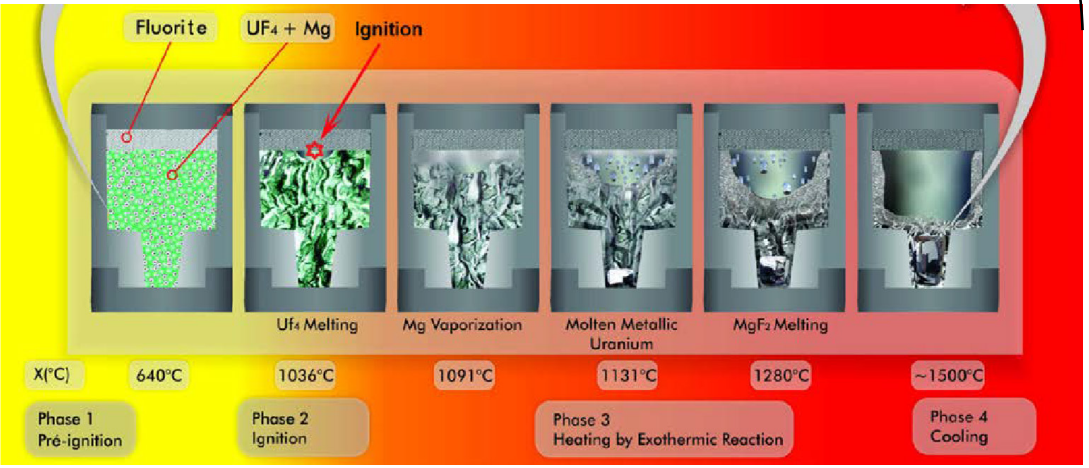
Refs.:
[1] National Research Council. 2000. Electrometallurgical Techniques for DOE Spent Fuel Treatment: Final Report. Washington, DC: The National Academies Press. https://doi.org/10.17226/9883.
[2] Fredrickson, G. L, et al. 2022. History and status of spent fuel treatment at the INL Fuel Conditioning Facility. Progress in Nuclear Energy 143, 104037, 2022.
[3] J.J. Laidler, et al.. Development of pyroprocessingtechnology. Progress in Nuclear Energy, 31(1):131–140, 1997.

F1: Fabrication of Metallic Fuel

- Based on US experience of SFR fuel manufacturing (EBR-I, EBR-II, FFTF)
- Reduction of enriched uranium to metal
  - Reduction of UF_4 or uranium oxides by metals (Ca, Mg, Al, Ba)
  - Electrolytic reduction of uranium oxide
- Alloying and casting to form the metallic slug
  - Most widely used: vacuum induction melting, alloying agent containing Pu and Zr
- Machining and thermo-mechanical processing to form metallic fuel pellet

Major hazards:

- Release of hazardous or corrosive chemicals
- Criticality from misfeeding or mishandling of fuel
- Release of radiological materials from leaking containers



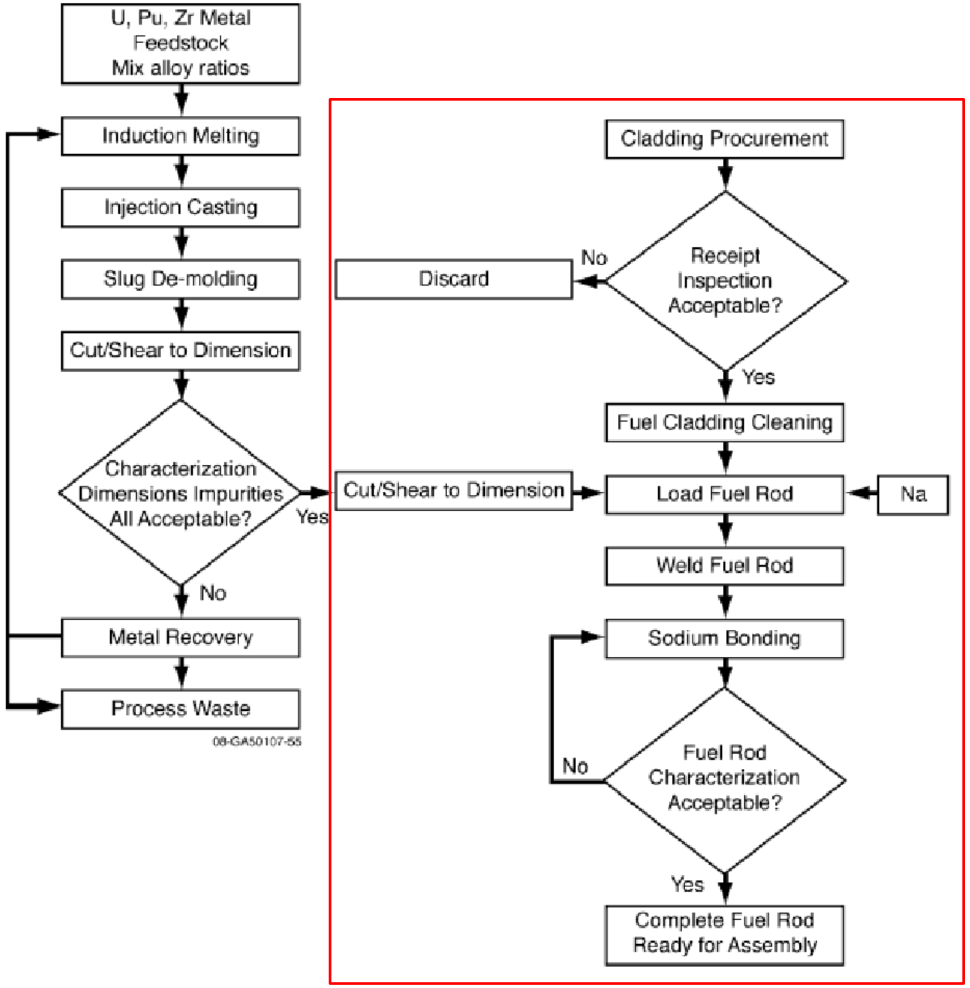
Ref.: N.L. LaHaye, D.E. Burkes, "Metal Fuel Fabrication Safety and Hazards - TO NRC-HQ-25-17-T-005, Non LWR LTD2," Pacific Northwest National Laboratory, PNNL-28622, 2019.

F2: Fabrication of Fuel Assemblies

1. Fuel rod fabrication:

- Fuel cladding tube is fabricated and cleaned
- Cladding tube is loaded with sodium to facilitate bonding
- Fuel slugs are loaded into the cladding tube
- Fuel cladding tube is closure welded to achieve sealing

2. Fuel assembly manufacturing



Major hazards:

- Release of hazardous or corrosive chemicals/gases
- Criticality from misfeeding or mishandling of fuel
- Release of radiological materials or sodium from rods

Ref.: D. E. Burkes, et al. A US Perspective on Fast Reactor Fuel Fabrication Technology and Experience Part 1: Metal Fuels and Assembly Design. Journal of Nuclear Materials, 389:458–469, 2009.

T2: Transportation of Fresh Fuel Assemblies to Plant

- SFR fuel have so far been transported in DOE-certified casks, but not in commercial size transportation packages
- Possible candidates: ES-3100 (used for transporting test reactor fuel) or other Type B shipping container
- ES-3100:
  - Certified for a variety of uranium bearing materials, including metals, with enrichments up to 100 wt.% ^{235}U.
  - Loading limits determined from enrichment, material form, and presence of spacers
  - Container length might limit SFR fuel type to be transported

Major hazards:

- Criticality due to water accidents and container drop
- Corrosion of sodium bond
- Reaction of sodium with water, air, or concrete in case of container ruptures

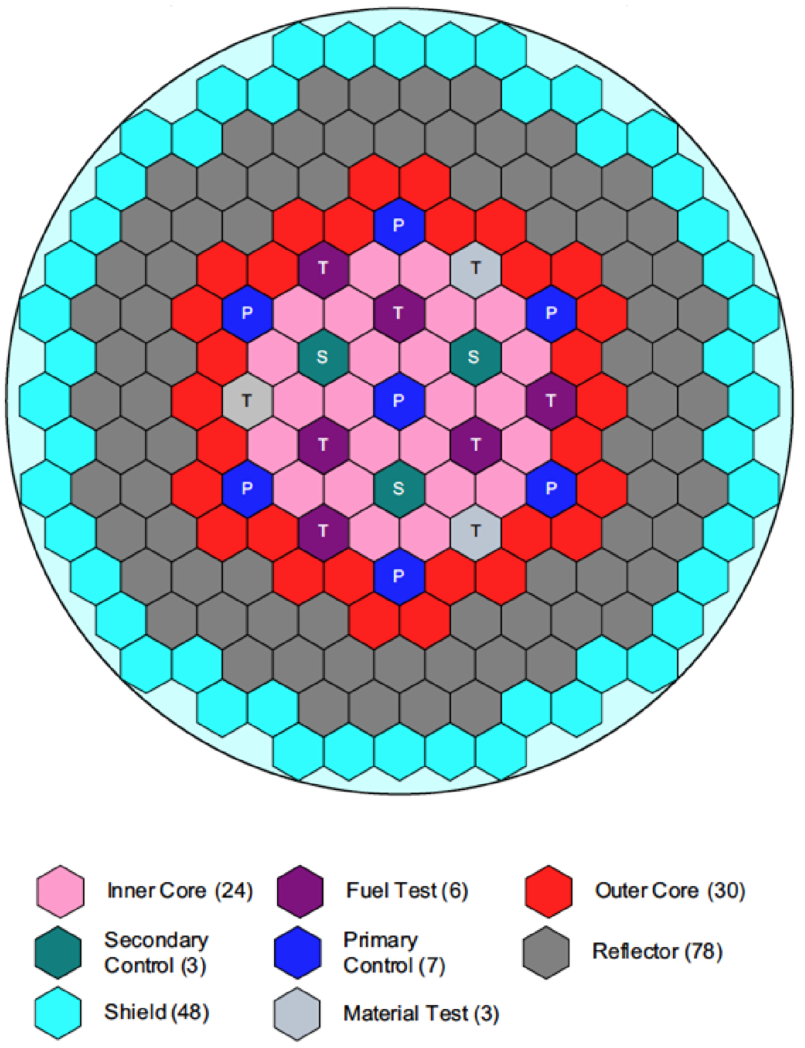
ES-3100

Ref.: J. Jarrell, "A Proposed Path Forward for Transportation of High-Assay Low-Enriched Uranium," INL Technical Report, INL/EXT-18-51518 Rev 0 (2018).

U1/U2/U4 – Utilization Stages

Ref.: Advanced Burner Test Reactor (ABTR)

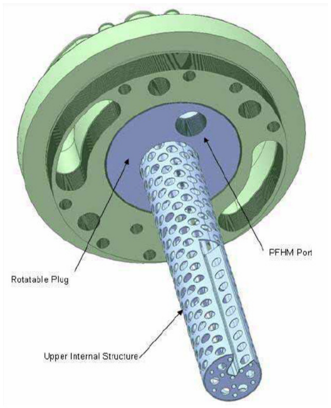
- Power: 250 MWt
- Fuel: metallic U/TRU-Zr
- Inner core assemblies:
  - 16.5% TRU fraction, 12 cycle lifetime, up to 94.5 GWd/tHM burnup
- Outer core assemblies:
  - 20.7% TRU fraction, 15 cycles lifetime, up to 92.6 GWd/tHM burnup
- Refueling for ~10 hours per assembly
- Operation for cycle time of 4 months followed by refueling of a maximum of 7 components:
  - 2 inner, 2 outer, 0-1 test, 0-1 control



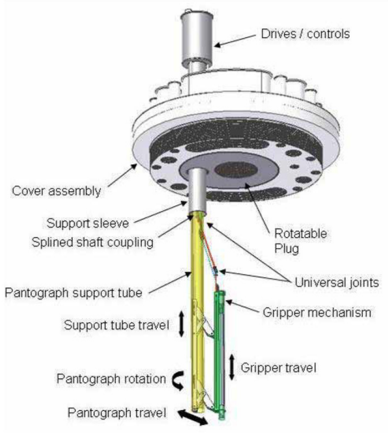
Ref.: Y. I. Chang, et al., "Advanced Burner Test Reactor Preconceptual Design Report," ANL-ABR-1/ANL-AFCI-173, Argonne National Laboratory, 2006.

U1/U2/U4: Major Components for Fuel Handling

- Pantograph fuel handling machine and rotatable plug: Transfer of fuel assemblies into the core, within core and into a storage rack, and from the core
- Storage rack: fresh and spent fuel assemblies, 36 positions
- Fuel unloading machine: inserting and retrieving core assemblies from the cue position on the storage rack; heating, cooling and inert gas atmosphere for transferring fuel assemblies between the core and an IBC
- Intra-building casks (IBC): lead-shielded inter-building casks with inert gas atmosphere, with or without active cooling
- Intra-building transfer tunnel: transfer of assemblies within inter-building cask

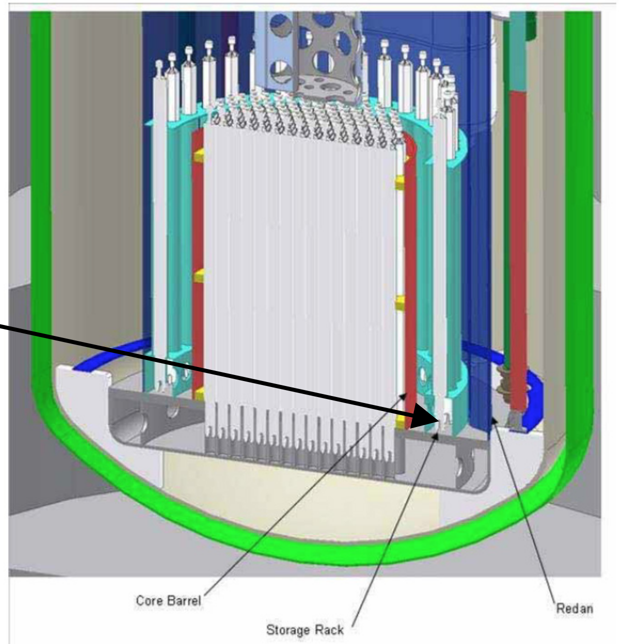


Rotatable plug



Pantograph

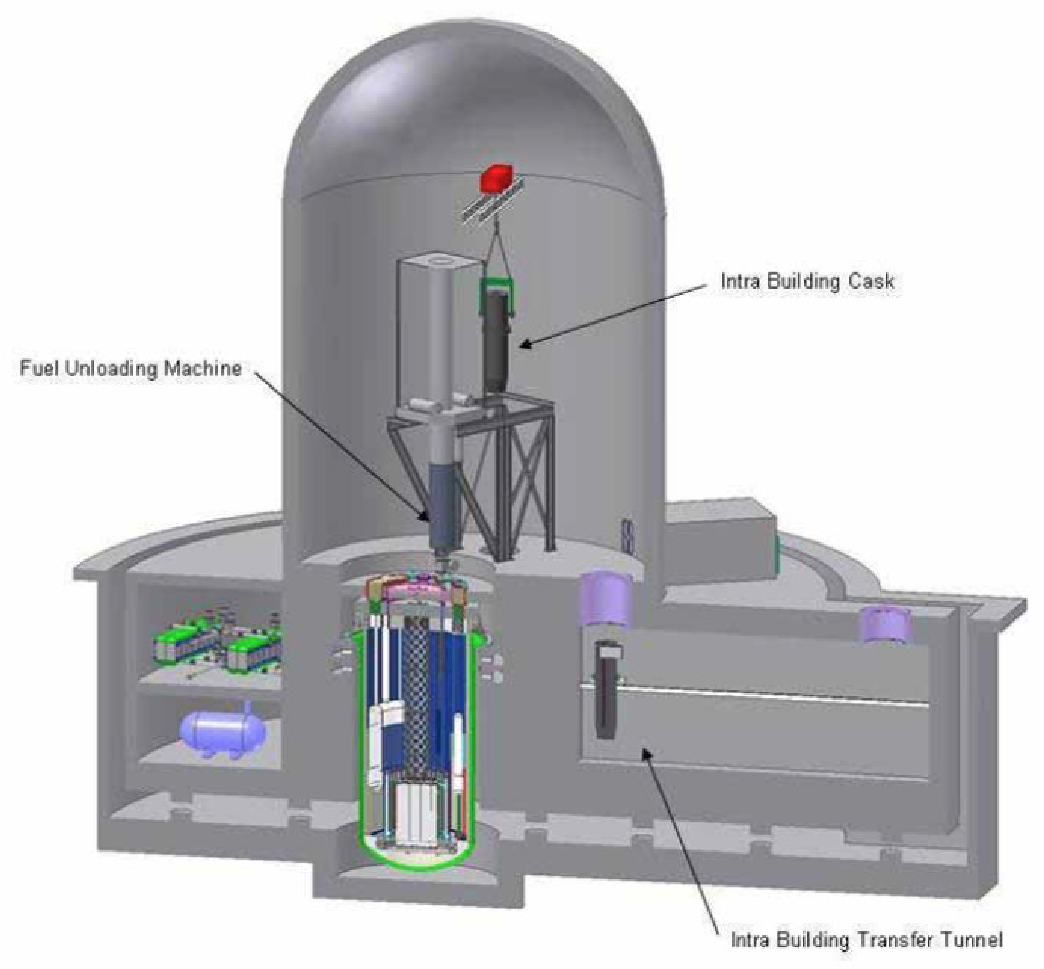
Storage rack



Ref.: Y. I. Chang, et al., "Advanced Burner Test Reactor Preconceptual Design Report," ANL-ABR-1/ANL-AFCI-173, Argonne National Laboratory, 2006.

U1/U2/U4: Major Hazards

Major hazards:
- Reaction of sodium with water, air, or concrete
- Corrosion of sodium bond
- Inadequate heat removal due to early removal of assembly from core or insufficient cooling by cask
- Damage to fuel assembly causing fission product release
- Criticality due to incorrect assembly pickup and drop off locations (consider sodium opaqueness)



Ref.: Y. I. Chang, et al., "Advanced Burner Test Reactor Preconceptual Design Report," ANL-ABR-1/ANL-AFCI-173, Argonne National Laboratory, 2006.

Summary

Major differences in the SFR fuel cycle compared to LWR:

- Use of U-Zr (HALEU) fuel, U/TRU-Zr fuel, and potentially reprocessed fuel
- No approved *commercial* size transportation and storage packages for SFR fuel assemblies with fresh fuel or reprocessed fuel
- New chemicals and processes for metallic fuel fabrication
- Use of sodium bond and sodium coolant
- Remote fuel handling and high reliance on I&C due to opaqueness of sodium coolant

Major identified hazards:

- Higher enrichment impacting criticality during UF_6 and fuel assembly storage and transportation
- Hazards from the use of the various chemicals (spills, reaction with water, fire, explosion)
- Sodium reaction with air and water, and sodium corrosion

Additional details needed:

- Fresh and spent fuel assembly storage details
- Detailed SFR containment and building design
- Details about specifications and operation of a reprocessing facility

Demonstration of SCALE for SFR Fuel Cycle Analysis

D. Hartanto

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OBJECTIVE AND APPLICATIONS

Objective: Demonstrate use of SCALE for simulating accident scenarios in all stages of the nuclear fuel cycle for Sodium-cooled Fast Reactors (SFR)

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.

Scenario 2: Criticality event / fissile material buildup during reprocessing (R1)

- Accident: Misfeed of material into the electro-processing batch leading to fissile material buildup / criticality as materials collect on the cathode.
- Analysis: Determine fuel inventory and perform SCALE criticality calculations.

Scenario 3: Release of fission products during reprocessing (R1)

- Accident: A leak in the waste stream storage tank allows for release of fission products during reprocessing.
- Analysis: Determine fuel inventory and perform SCALE activity calculations.

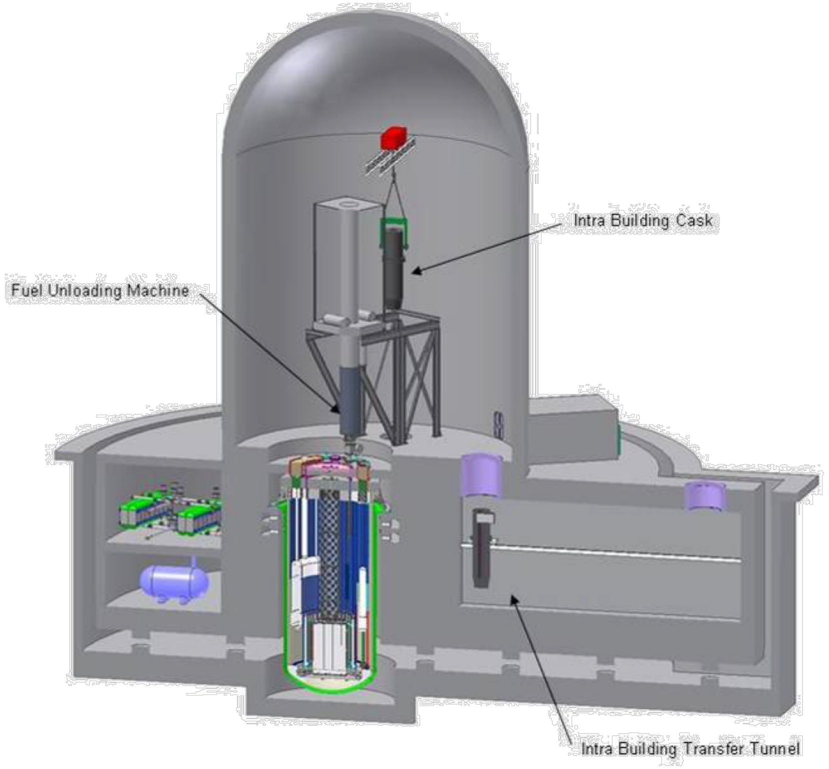

Figure II.7-1 Schematic of Fuel Handling System

ABTR reactor building

Ref.: Chang, Y. I., et al. Advanced Burner Test Reactor – Preconceptual Design Report. Technical Report ANL-ABR-1 (ANL-AFCI-173), Argonne National Laboratory, 2006.

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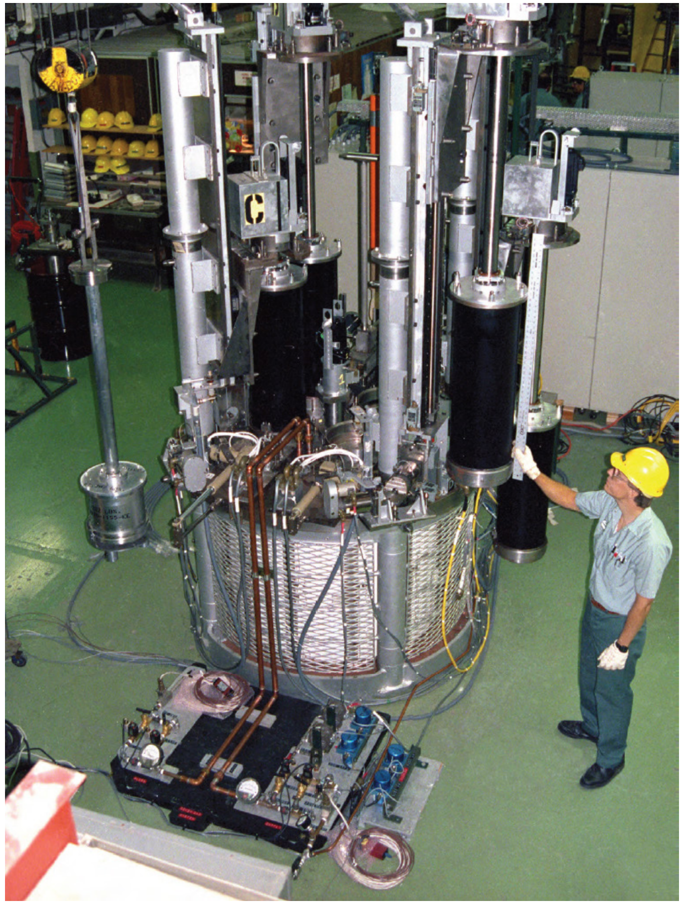


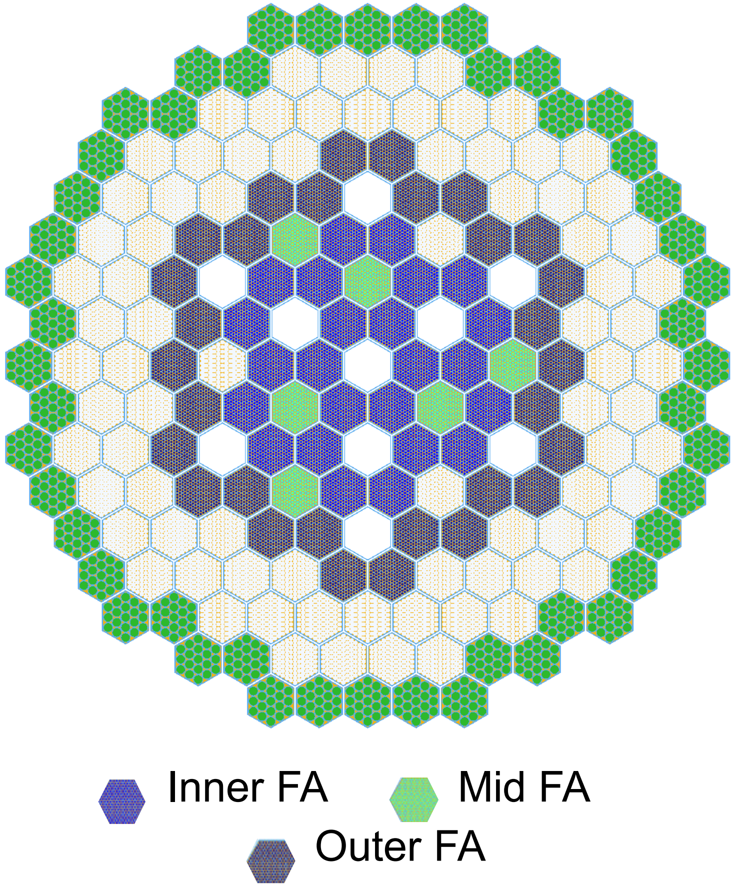
Electrorefiner

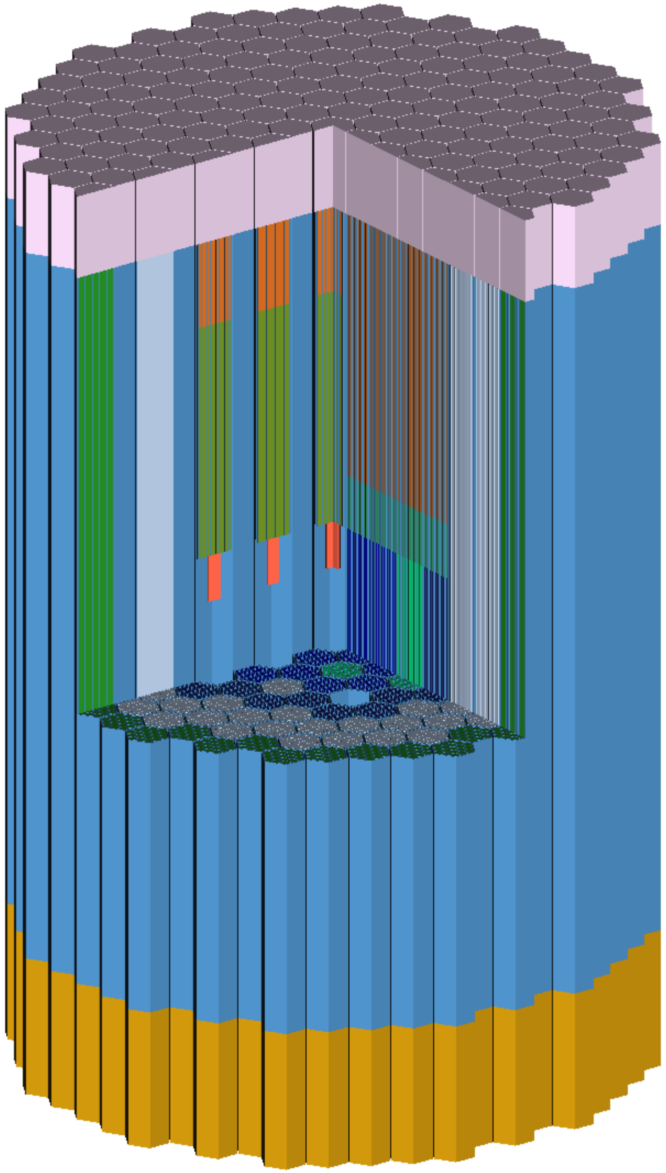
Ref.: Pyroprocessing Technologies Brochure, Argonne National Laboratory

REFERENCE SODIUM FAST REACTOR DESIGN

Advanced Burner Test Reactor (ABTR)

Reactor Power	250 MW_t, 95 MWe
Coolant Temperature	355°C/510°C
Fuel	Metallic
Cladding and Duct	HT-9
Cycle Length	4 months





SCALE ABTR Model

Refs.:

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

[2] Kim, T. K. Benchmark Specification of Advanced Burner Test Reactor. Technical Report ANL/NSE-20/65, Argonne National Laboratory, 2020.

APPLIED SCALE6.3.1 SEQUENCES

Rapid inventory generation with **ORIGAMI**

- Depletion and decay solver (ORIGEN)
- Requires pre-calculated ORIGEN cross-section libraries (generated in previous work for the ABTR*)
- **Output**:
  - Nuclide inventory of irradiated fuel
  - Decay heat and activity of irradiated fuel
  - Photon and neutron source terms of irradiated fuel
  - Activation sources of irradiated non-fuel materials (Zr, HT9, and SS316)

Shielding & radiation dose calculations with **MAVRIC**

- Monte Carlo photon and neutron transport code (MONACO) with automated variance reduction for shielding analyses
- Requires radiation source terms.
- **Output**:
  - Spatial flux/dose rate distributions

Criticality calculation with **CSAS**

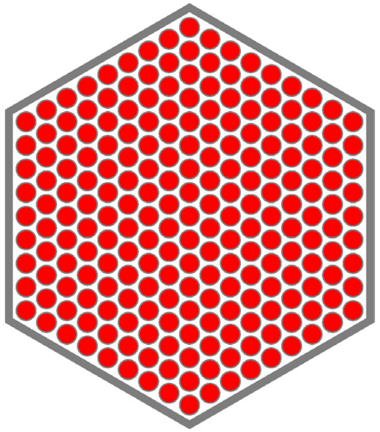
- Monte Carlo neutron transport code (KENO or Shift) for criticality safety analysis
- **Output:**
  - Multiplication factor
  - Spatial flux and fission density distributions

Nuclide inventory and decay heat of the irradiated fuel are passed to MELCOR.

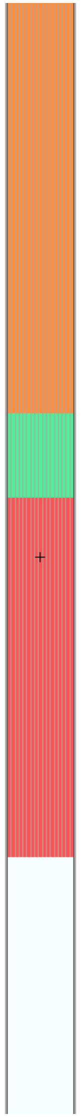
Ref:
[1] Wieselquist, W. A., Lefebvre, R. A., Eds., SCALE 6.3.1 User Manual, ORNL/TM-SCALE-6.3.1, Oak Ridge National Laboratory, 2023.
*[2] *Shaw, A, et al. SCALE Modeling of the Sodium Cooled Fast-Spectrum Advanced Burner Test Reactor. Technical Report ORNL/TM-2022/2758, Oak Ridge National Laboratory, 2022.*

SPENT NUCLEAR FUEL

ABTR TRU Fuels	**Source terms for all scenarios (ABTR TRU Inner)**
	U/TRU-10Zr
	16.5 wt.% (inner) & 20.7 wt.% TRU (outer)
	Specific power: 65.6 GW/tHM (inner) & 51.4 GW/tHM (outer)
	Discharged BU: 94.5 GWd/tHM (inner) & 92.6 GWd/tHM (outer)
ABTR HALEU Fuel	**Source terms for all scenarios**
	U-10Zr
	16.5 wt.% U-235
	Specific power: 46.2 GW/tHM
	Discharged BU: 149.74 GWd/tHM
PWR Fuel	**Source terms for scenarios 2 and 3**
	UO_2
	4.95 wt.% U-235
	Specific power: 33.7 GW/tHM
	Discharged BU: 50.00 GWd/tHM



ABTR fuel assembly



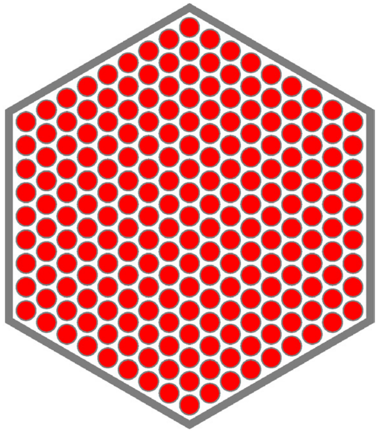
Refs.:
[1] Kim, T. K. Benchmark Specification of Advanced Burner Test Reactor. Technical Report ANL/NSE-20/65, Argonne National Laboratory, 2020.
[2] Natrium Clearpath Webinar (nationalacademies.org).
[3] Kim, T. K. and T. A. Taiwo, Fuel Cycle Analysis of Once-Through Nuclear Systems. Technical Report ANL-FCRD-308, Argonne National Laboratory, 2010.

SPENT NUCLEAR FUEL

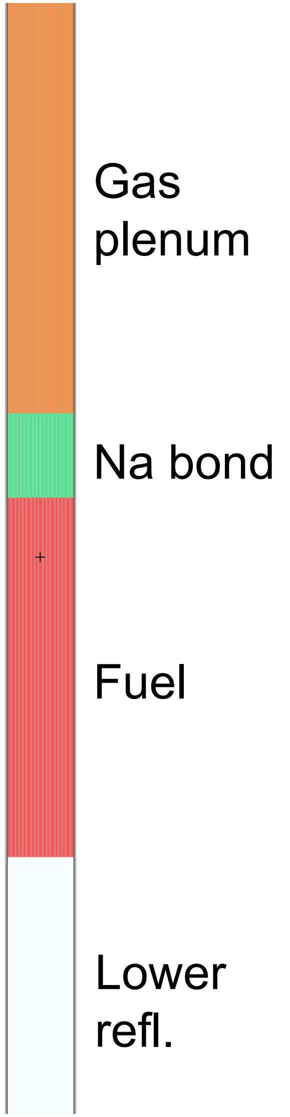
Rapid inventory generation with ORIGAMI

Irradiation history:

- TRU Inner
  - Loaded for 12 cycles
  - 120 days per cycle
- TRU Outer
  - Loaded for 15 cycles
  - 120 days per cycle
- HALEU
  - Loaded for 6 cycles
  - 540 days per cycle
- Assuming 10 days of cooling time between cycles
- Discharged fuel assembly is planned to be stored for 7 reactor cycles in the in-vessel storage (IVS)



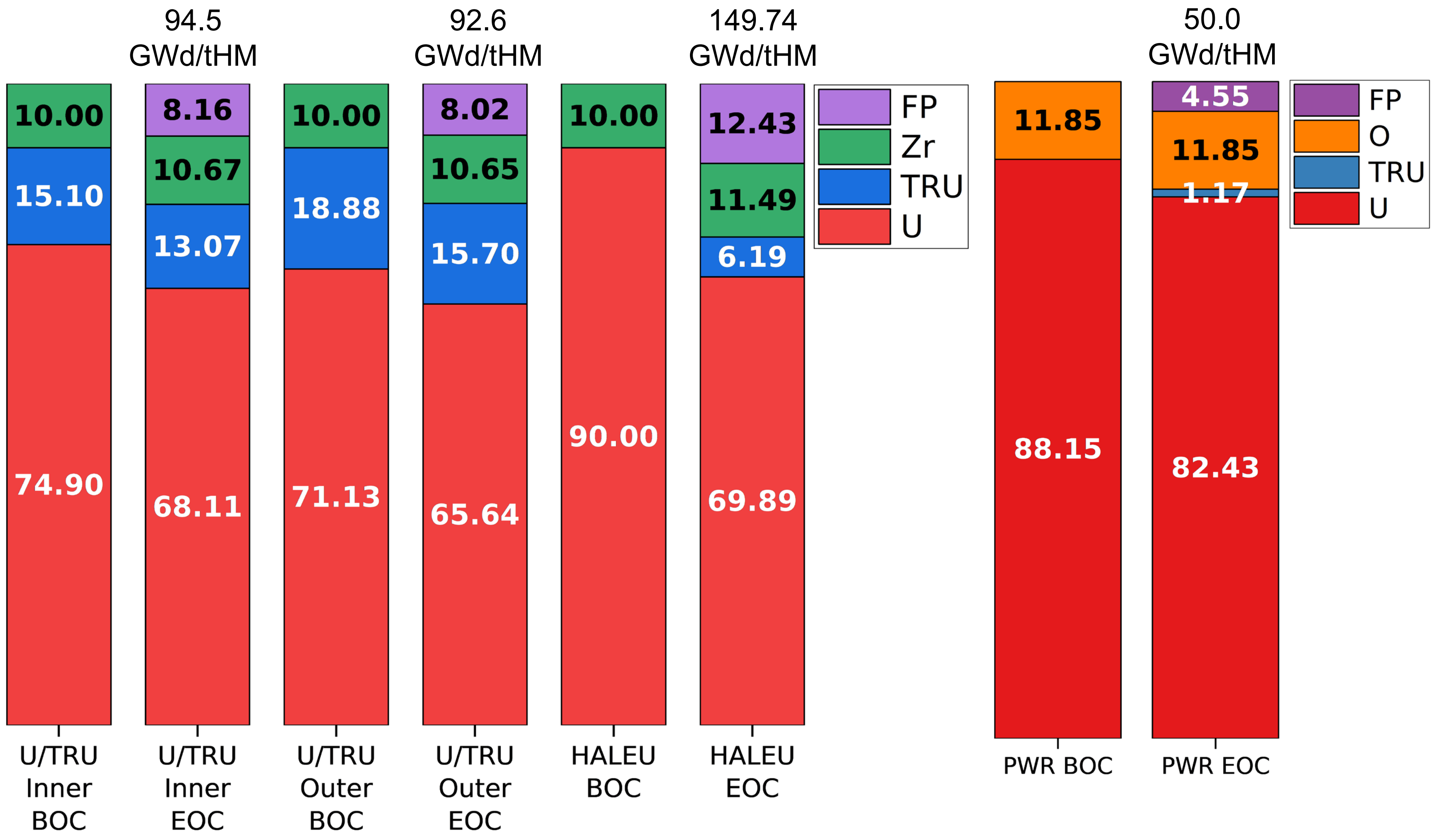
ABTR
fuel assembly



SPENT NUCLEAR FUEL - COMPOSITION

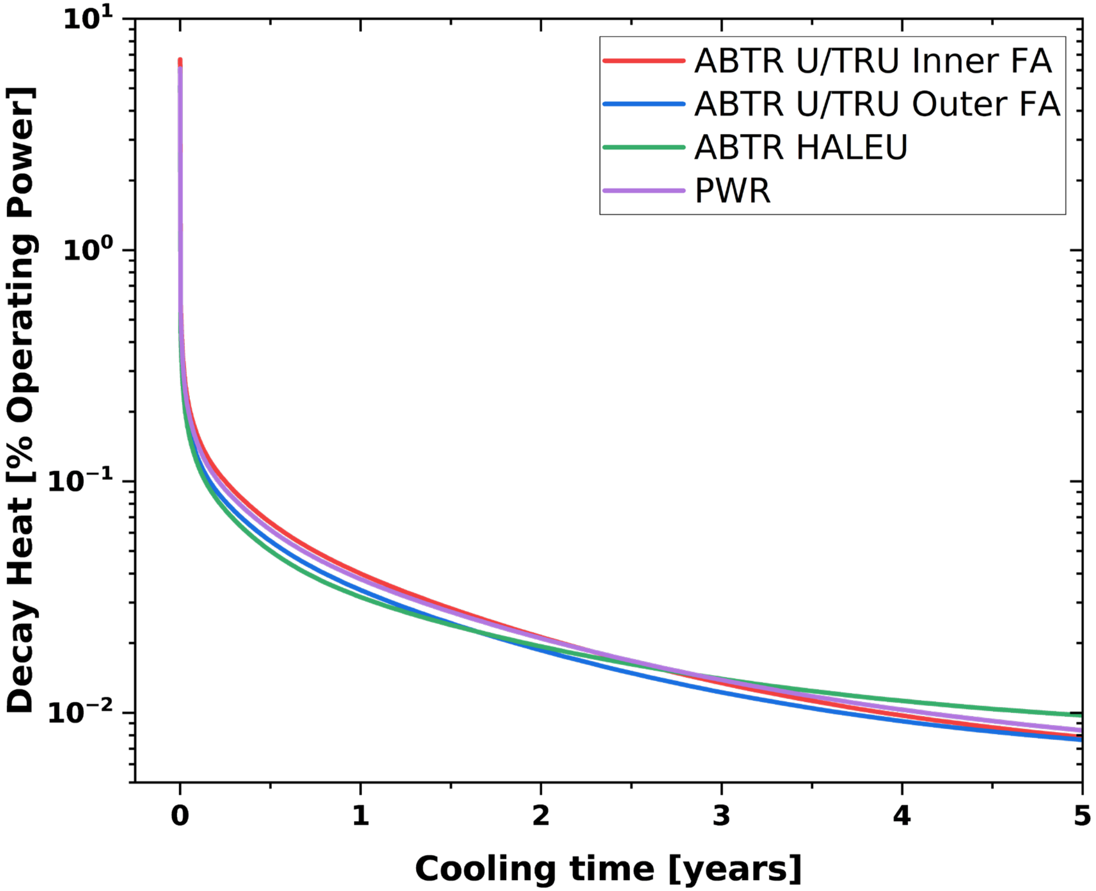
Composition distribution in the fuels at BOC and EOC (wt.%)

- Since all ABTR fuels have a higher burnup, they produce more TRUs and FPs than the PWR's.
- More FPs are produced by ABTR HALEU fuel than U/TRU fuel due to higher burnup (~150 GWd/tHM).
- ABTR U/TRU fuels have higher TRU fraction at EOC compared to the HALEU fuel.



BOC: beginning of cycle
EOC: end of cycle
FP: fission product
TRU: transuranics

SPENT NUCLEAR FUEL – DECAY HEAT

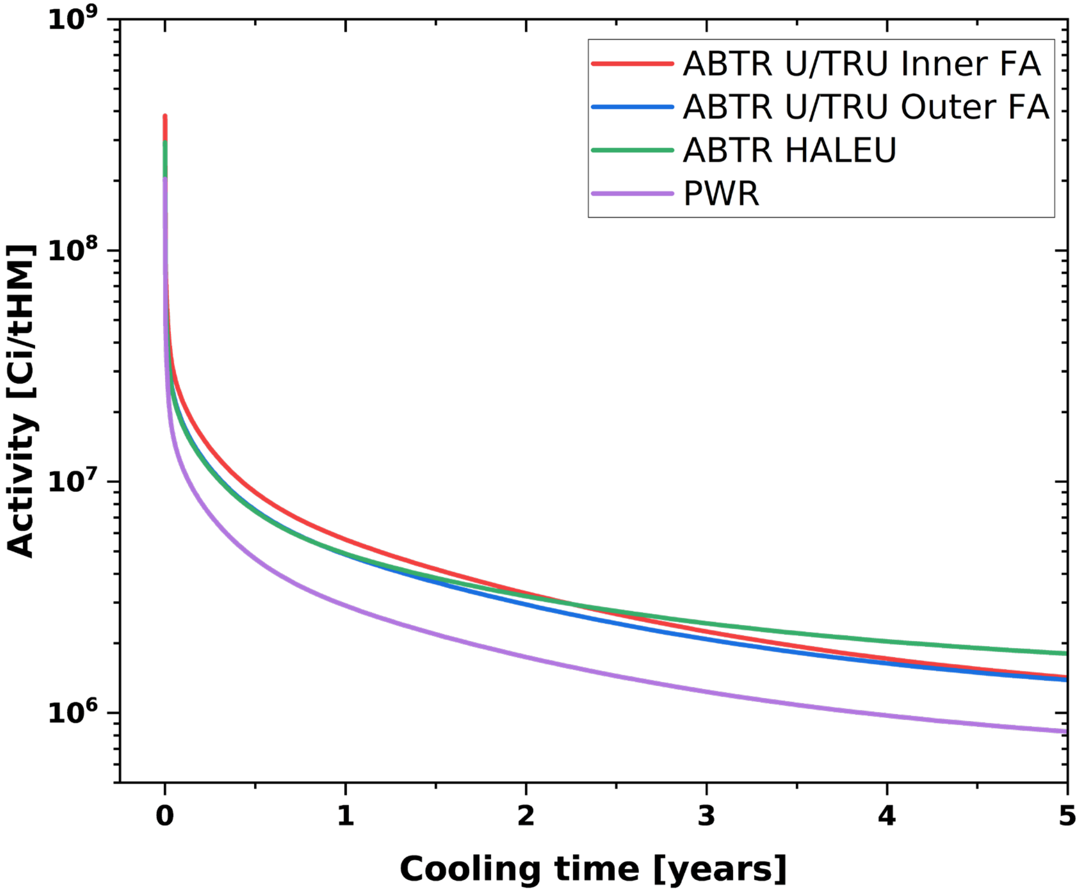


- Decay heat at shutdown is similar between the different fuel types (~5-7% power)
- Initially, slightly higher for the U/TRU inner fuel due to higher specific power although its burnup is lower than HALEU

Top 5 decay heat contributors at 10 days and 5 years (ABTR) and *10 years (PWR)

Fuel	At 10 days of cooling time	At 5 years of cooling time
U/TRU Inner	^{140}La (21%) ^{106}Rh (12%) ^{144}Pr (9%) ^{95}Nb (8%) ^{95}Zr (8%)	^{137m}Ba (22%) ^{106}Rh (14%) ^{90}Y (12%) ^{238}Pu (9%) ^{134}Cs (7%)
U/TRU Outer	^{140}La (21%) ^{106}Rh (12%) ^{144}Pr (9%) ^{95}Nb (8%) ^{95}Zr (8%)	^{137m}Ba (22%) ^{106}Rh (12%) ^{90}Y (12%) ^{238}Pu (10%) ^{134}Cs (6%)
HALEU	^{140}La (21%) ^{144}Pr (11%) ^{95}Nb (9%) ^{95}Zr (9%) ^{106}Rh (7%)	^{90}Y (29%) ^{137m}Ba (29%) ^{134}Cs (11%) ^{137}Cs (7%) ^{238}Pu (6%)
PWR*	^{140}La (21%) ^{144}Pr (10%) ^{95}Nb (8%) ^{106}Rh (8%) ^{95}Zr (8%)	^{90}Y (25%) ^{137m}Ba (25%) ^{238}Pu (11%) ^{244}Cm (11%) ^{137}Cs (7%)

SPENT NUCLEAR FUEL - ACTIVITY



- Similar trends compared to decay heat
- PWR has the lowest activity due to lower FPs built-up

Top 5 activity contributors at 10 days and 5 years (ABTR) and *10 years (PWR)

Fuel	At 10 days of cooling time	At 5 years of cooling time
U/TRU Inner	^{103}Ru (8%) ^{103m}Rh (8%) ^{95}Nb (7%) ^{95}Zr (7%) ^{141}Ce (6%)	^{137}Cs (19%) ^{137m}Ba (18%) ^{241}Pu (15%) ^{147}Pm (12%) ^{90}Y (7%)
U/TRU Outer	^{103}Ru (8%) ^{103m}Rh (8%) ^{95}Nb (7%) ^{95}Zr (6%) ^{141}Ce (6%)	^{137}Cs (19%) ^{241}Pu (19%) ^{137m}Ba (18%) ^{147}Pm (11%) ^{90}Y (7%)
HALEU	^{95}Nb (8%) ^{95}Zr (7%) ^{103}Ru (6%) ^{103m}Rh (6%) ^{141}Ce (6%)	^{137}Cs (22%) ^{137m}Ba (21%) ^{90}Y (16%) ^{90}Sr (16%) ^{147}Pm (10%)
PWR*	^{140}La (32%) ^{95}Nb (14%) ^{95}Zr (13%) ^{103}Ru (9%) ^{134}Cs (6%)	^{137m}Ba (76%) ^{134}Cs (16%) ^{154}Eu (7%) ^{125}Sb (0.5%) ^{106}Rh (0.2%)

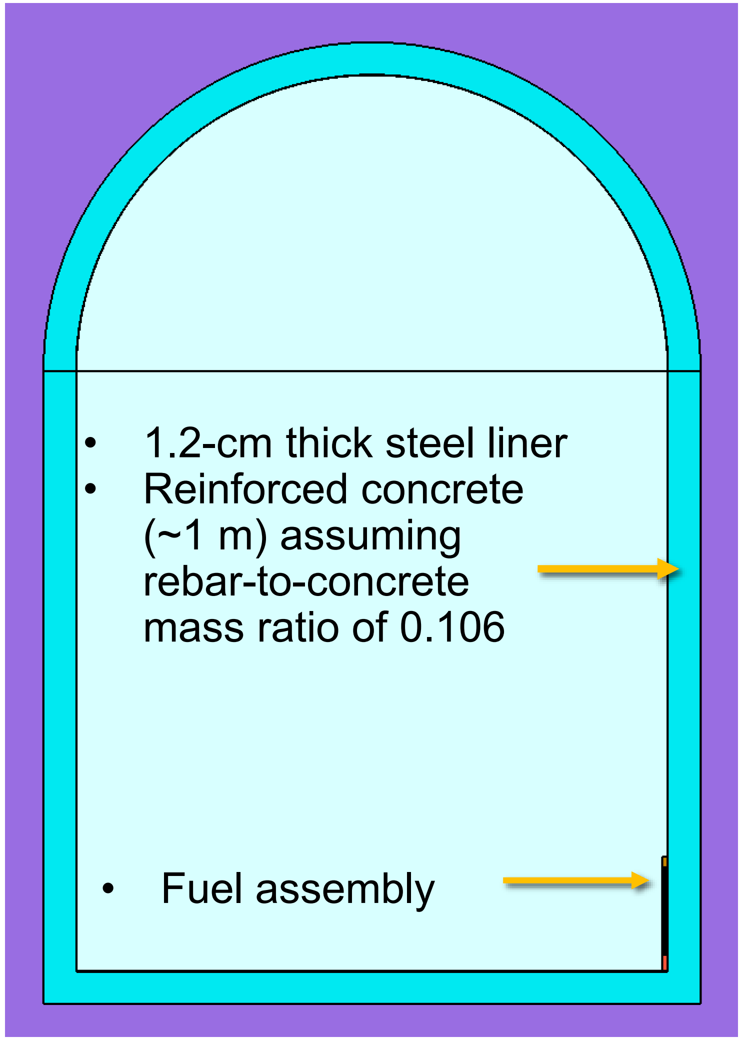
Scenario 1

Seismic event causing the refueling machine to fall and release the fuel assembly

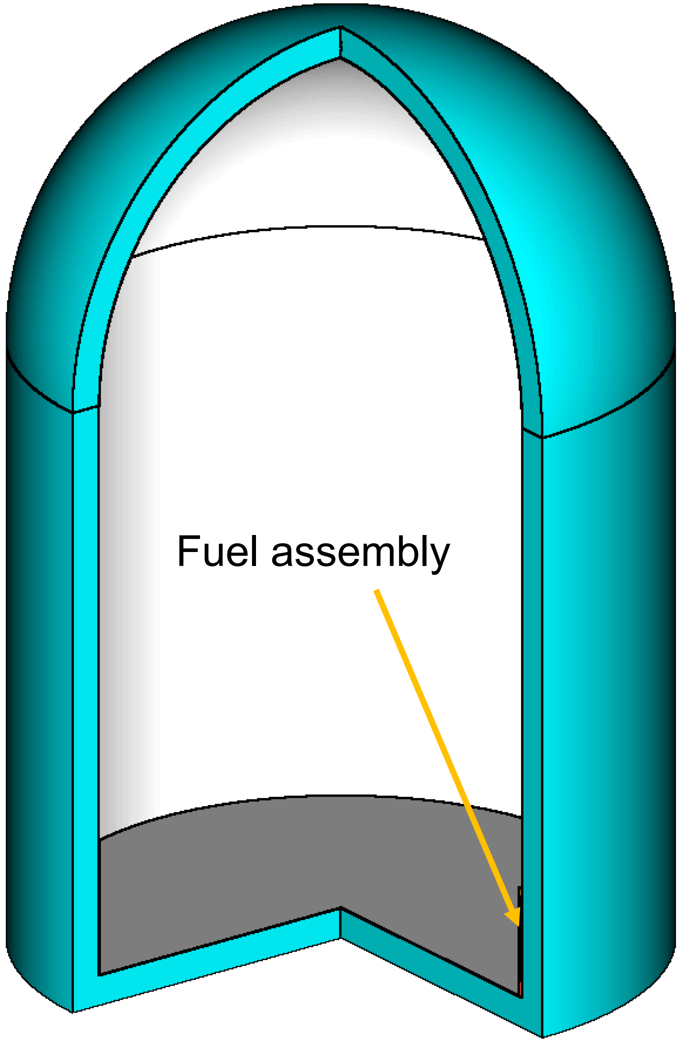
CONTAINMENT BUILDING (CB) MODEL

- Fuel assembly falls down from the refueling machine cask.
  - **ABTR HALEU** and **U/TRU** (Inner)
  - Case 1: Fuel assembly is cooled for **10 days**
  - Case 2: Fuel assembly is cooled for **7 reactor cycles**

- Radiation dose rate inside and outside of containment are calculated with MAVRIC using intact fuel assembly as radiation source (irradiated fuel and activation products).
  - ANSI standard (1977) flux-to-dose-rate factors
  - Cartesian and cylindrical mesh for dose calculations
  - Statistical error < 0.5%



MAVRIC model of the CB and unshielded fuel assembly (front view)



3D view of the CB with front quarter segment removed

Refs.:
[1] P. F. Peterson et al., Metal and Concrete Inputs for Several Nuclear Power Plants, Report UCBTH-05-001, 2005.
[2] Chang, Y. I., et al. Advanced Burner Test Reactor – Preconceptual Design Report. Technical Report ANL-ABR-1 (ANL-AFCI-173), Argonne National Laboratory, 2006.

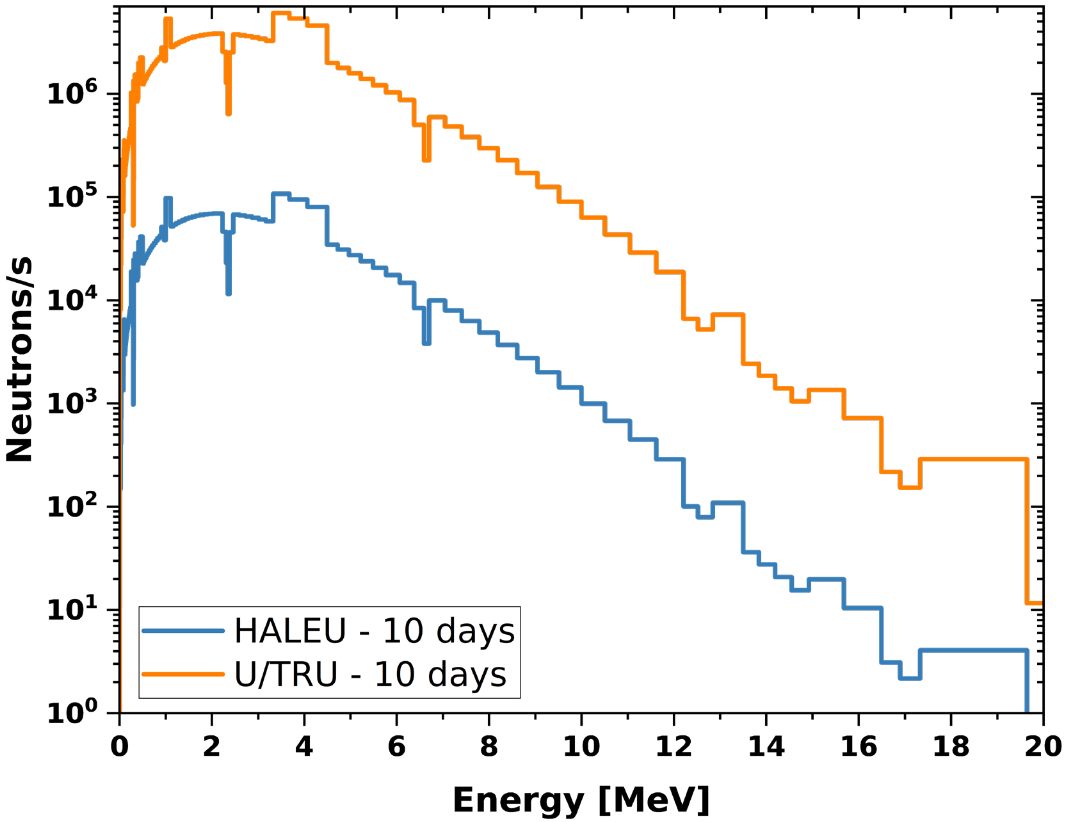
NEUTRON SOURCE TERMS

- Neutron sources from spontaneous fission
  - Fuel light element impurities might contribute additional neutron sources

Cooling time	ABTR HALEU	ABTR U/TRU (inner)
10 days	Cm-242 (74.2%) Pu-240 (17.3%)	Cm-242 (44.3%) Cm-244 (54.0%)
7 cycles	Pu-240 (71.3%) Pu-238 (16.2%) Cm-244 (11.5%)	Cm-244 (94.1%) Cm-242 (0.27%)

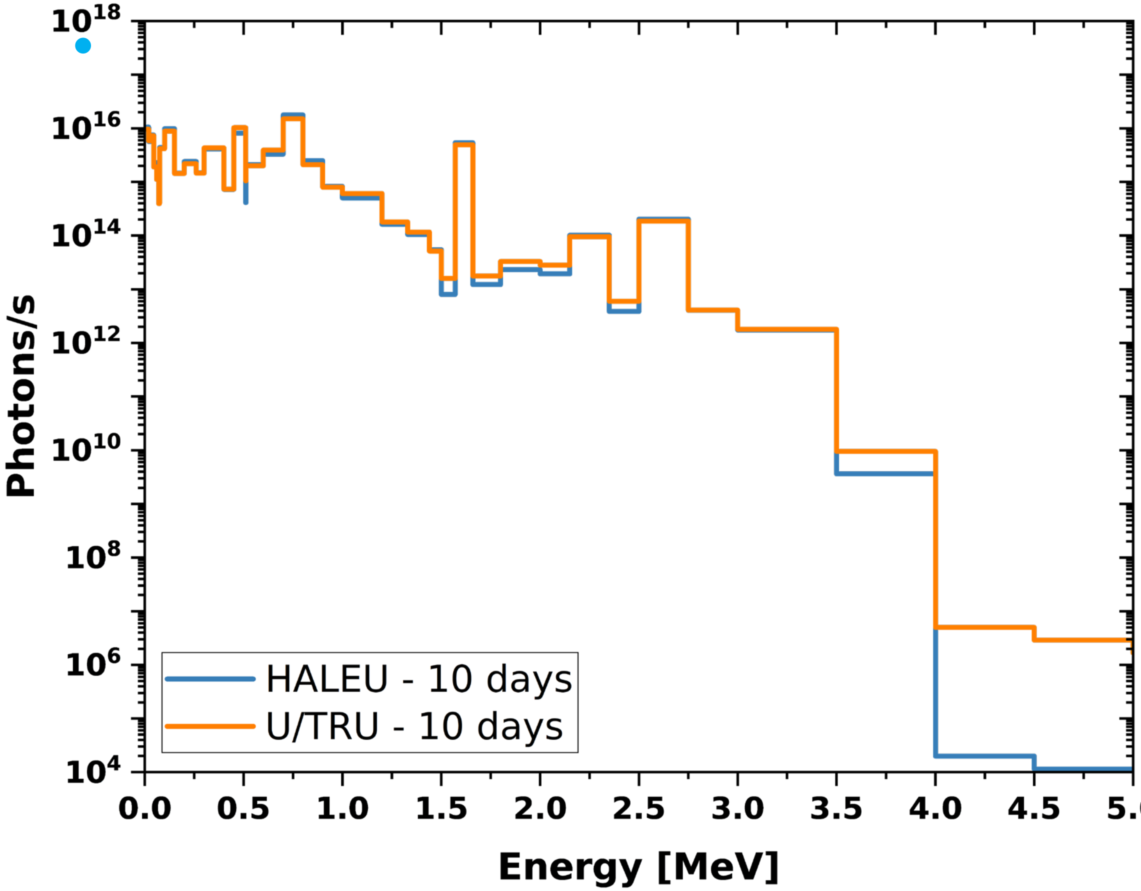
Half life:
Cm-242: 162.8 d
Cm-244: 18.10 y

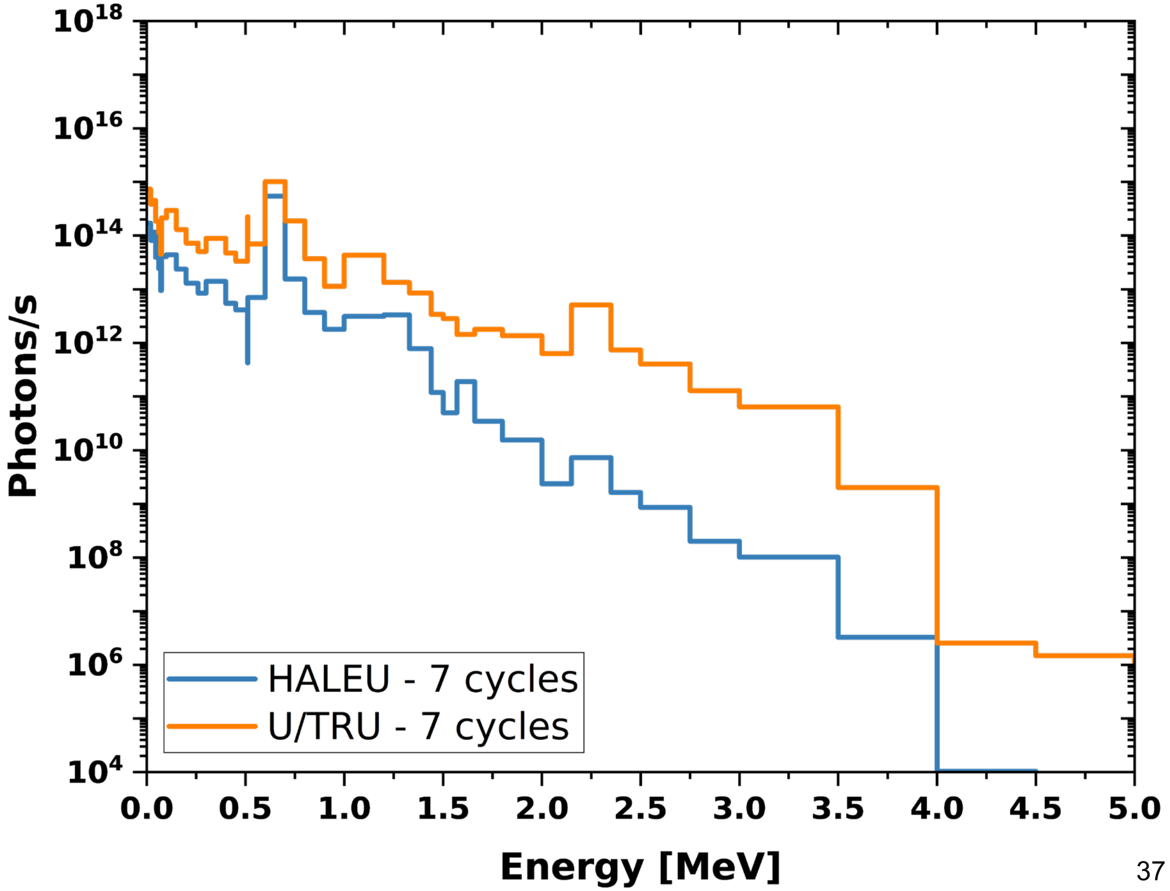
7 cycles of cooling time:
U/TRU: 840 d
HALEU: 3780 d



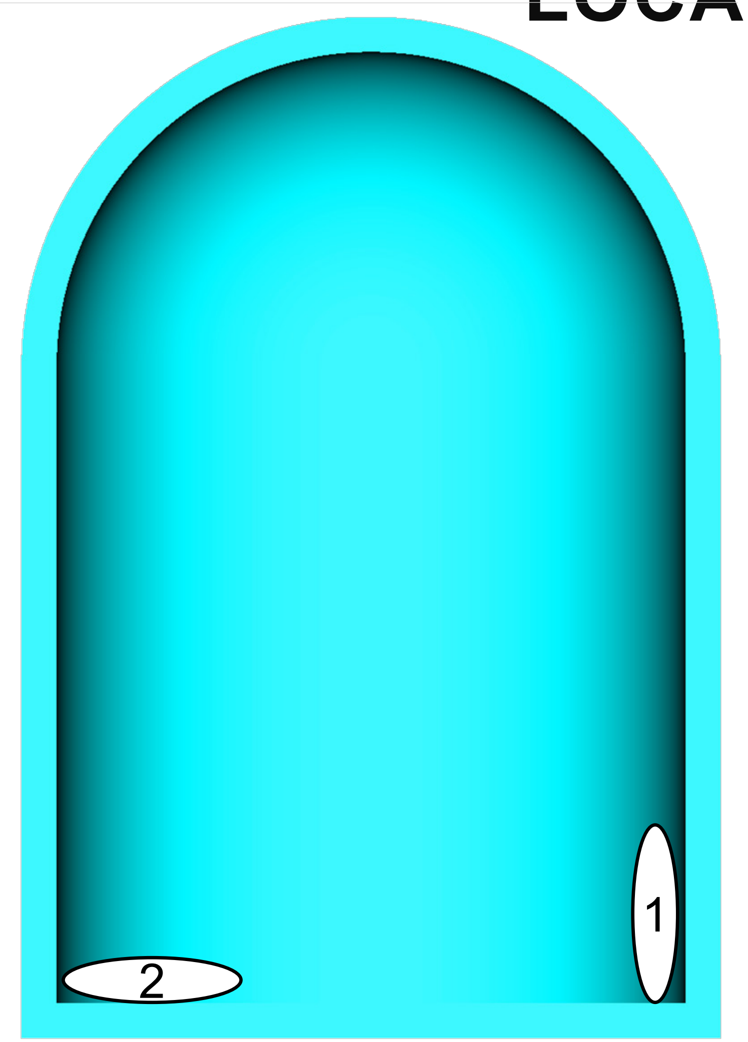
GAMMA SOURCE TERMS

- Strong fuel gamma radiation sources
- Total dose rate dominated by fuel gamma dose rate
- The neutron dose rate negligible as compared to the gamma dose rate (~6 orders of magnitude lower)

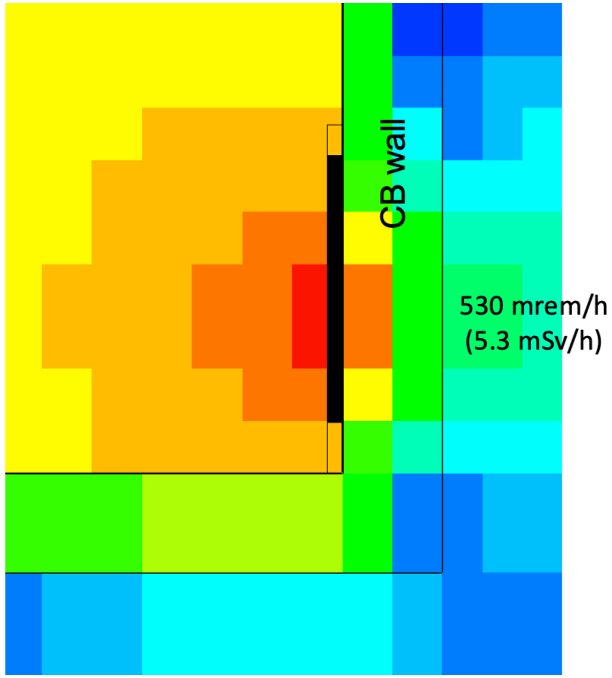




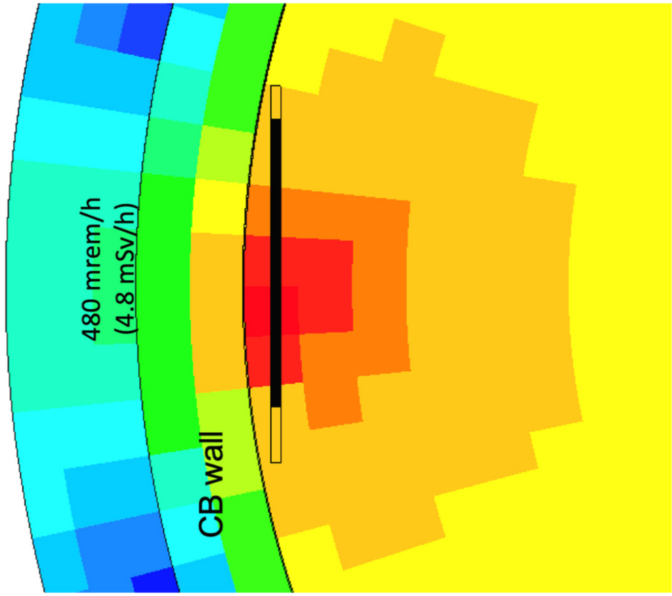
SENSITIVITY OF DOSE RATE TO FUEL ASSEMBLY LOCATION AND ORIENTATION



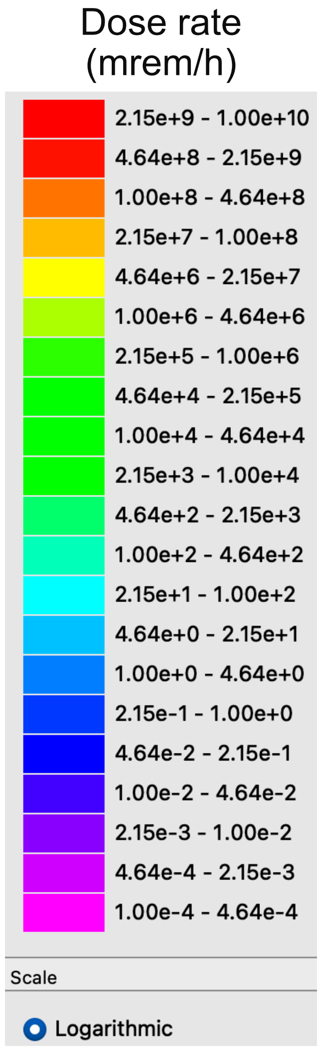
Location of the FA



Position 1 ABTR HALEU fuel assembly leaning on the containment wall (front view)



Position 2 ABTR HALEU fuel assembly lying on the floor next to containment wall (top view)



- Highest dose rate observed when fuel assembly leans on containment wall
  → This model is used for all dose rate calculations

MAIN BETA AND GAMMA EMITTERS

Nuclides important to the gamma source terms for both ABTR U/TRU and HALEU fuels

- 10 days of cooling

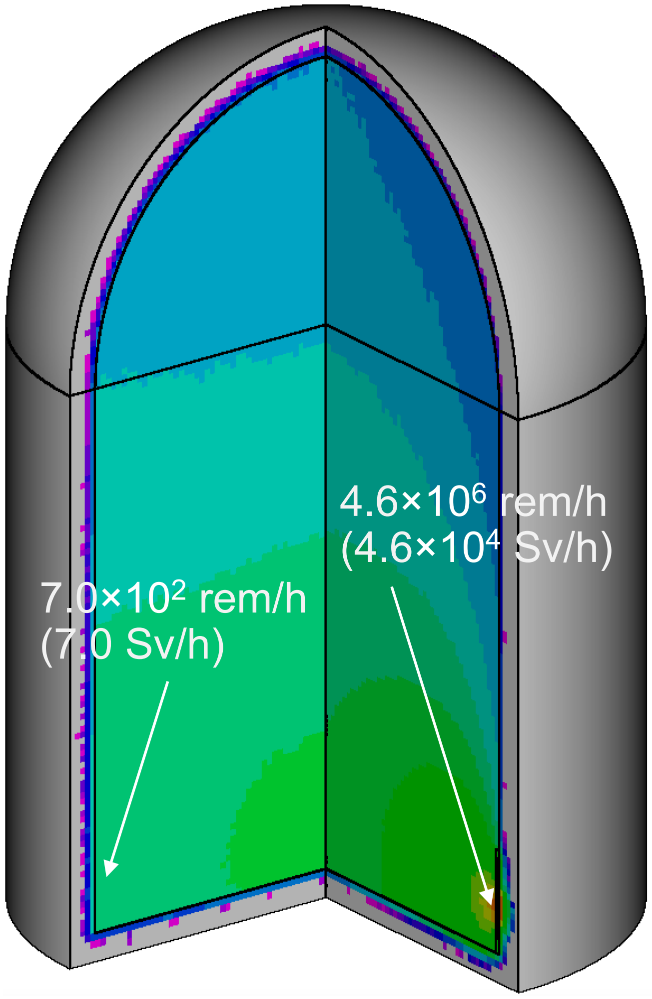
Nuclide	Half-life	Nuclide	Half-life
Y-91	58.5 d	Cs-137/Ba-137m	30.07 yr/2.552 m
Zr-95	64.02 d	Ba-140	12.75 d
Nb-95	34.99 d	La-140	1.678 d
Ru-103	39.27 d	Ce-144/Pr-144	284.6 d/17.28 m
Ru-106/Rh-106	1.02 yr/2.18 h	Nd-147	10.98 d
Sb-124	60.2 d	Pm-148m	42.3 d
Te-132/I-132	3.2 d/2.28 h	Eu-154	8.593 yr
Cs-134	2.065 yr	Eu-156	15.2 d
Cs-136/Ba-136m	13.16 d/0.308 s		

- 7 cycles of cooling

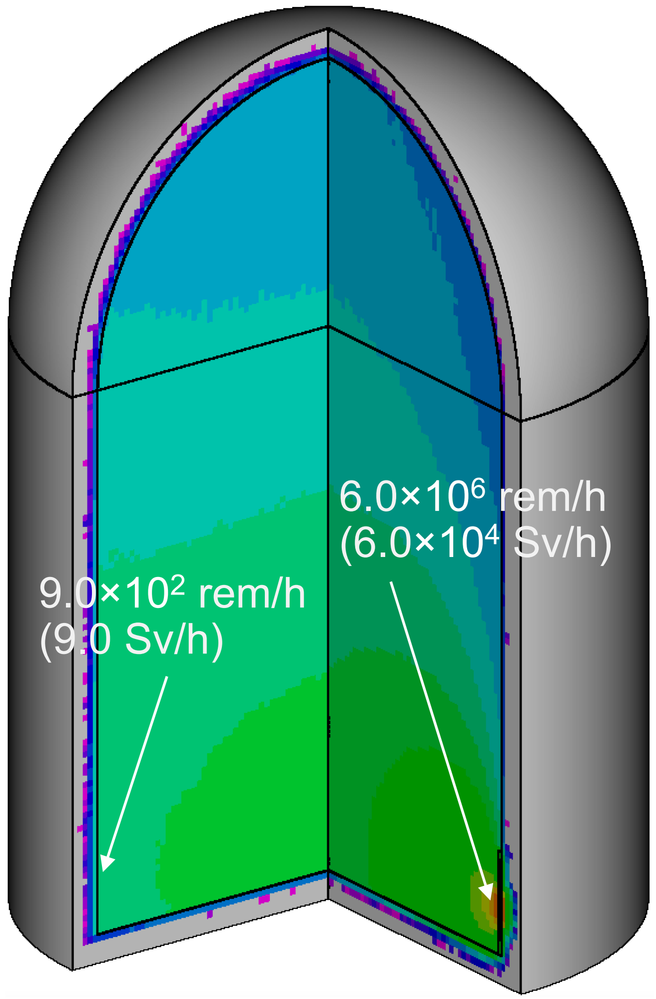
Nuclide	Half-life
Sr-90/Y-90	28.78 yr/2.67 d
Ru-106/Rh-106	1.02 yr/2.18 h
Ag-110m	249.8 d
Sb-125	2.758 yr
Cs-134	2.065 yr
Cs-137/Ba-137m	30.07 yr/2.552 m
Ce-144/Pr-144	284.6 d/17.28 m
Eu-152	13.54 yr
Eu-154	8.593 yr

DOSE RATE MAP INSIDE CB

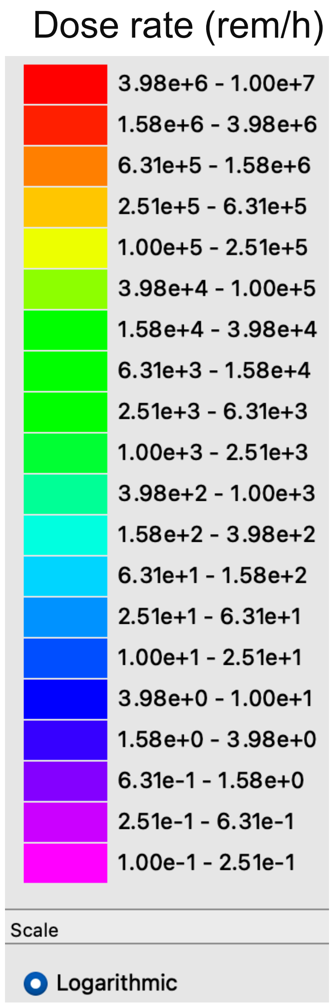
10 days cooling time



ABTR HALEU

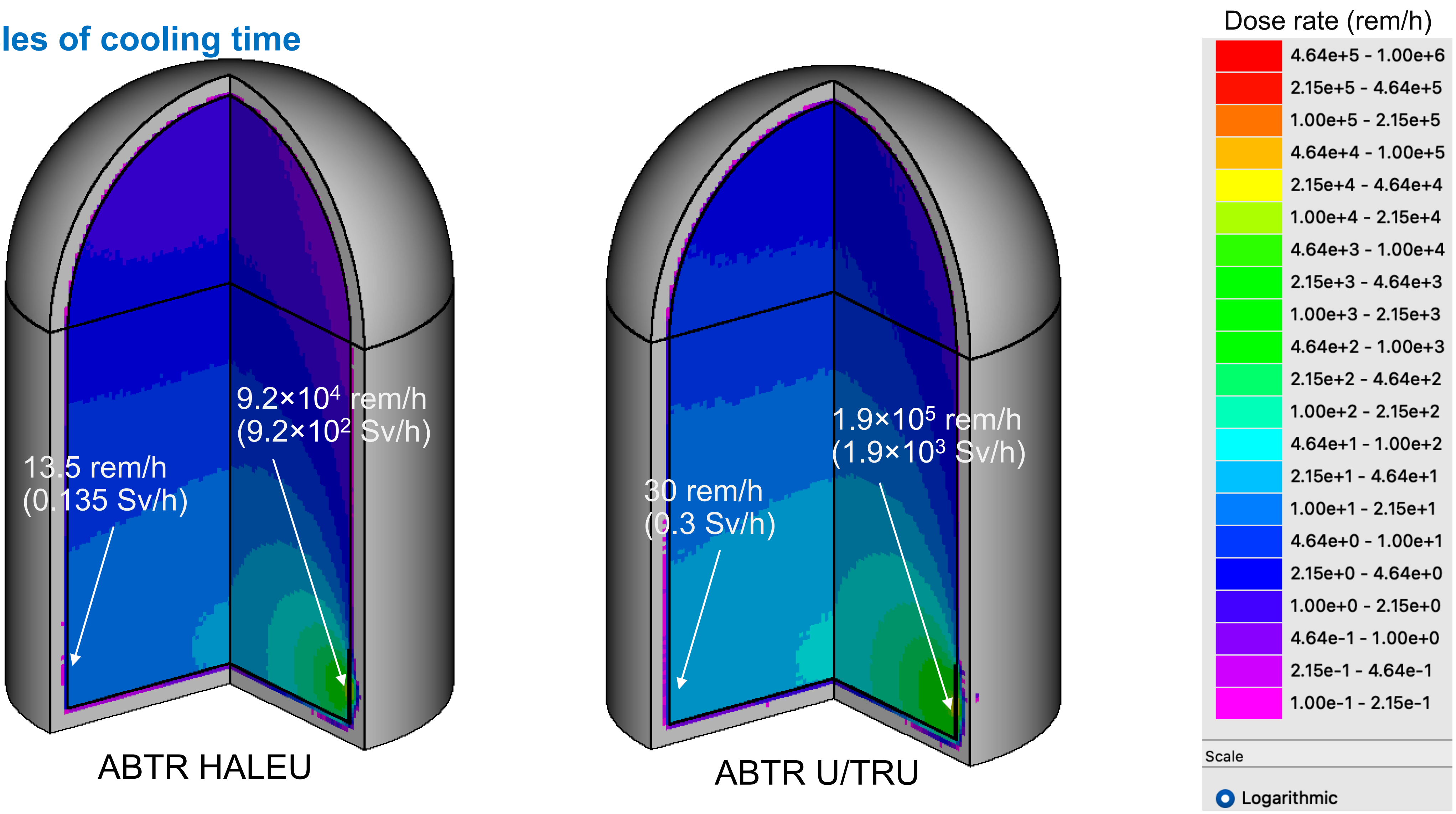


ABTR U/TRU



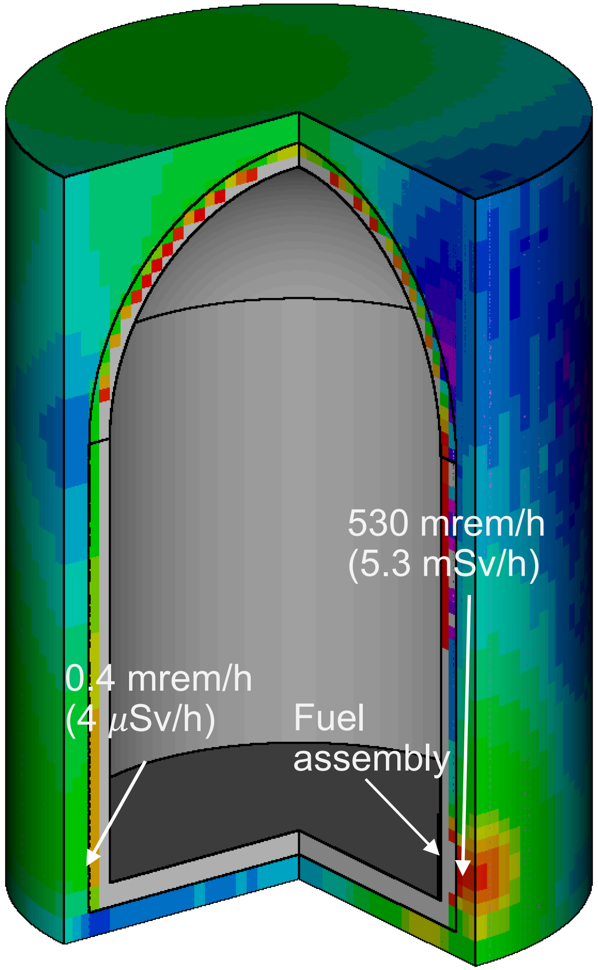
DOSE RATE MAP INSIDE CB

7 cycles of cooling time

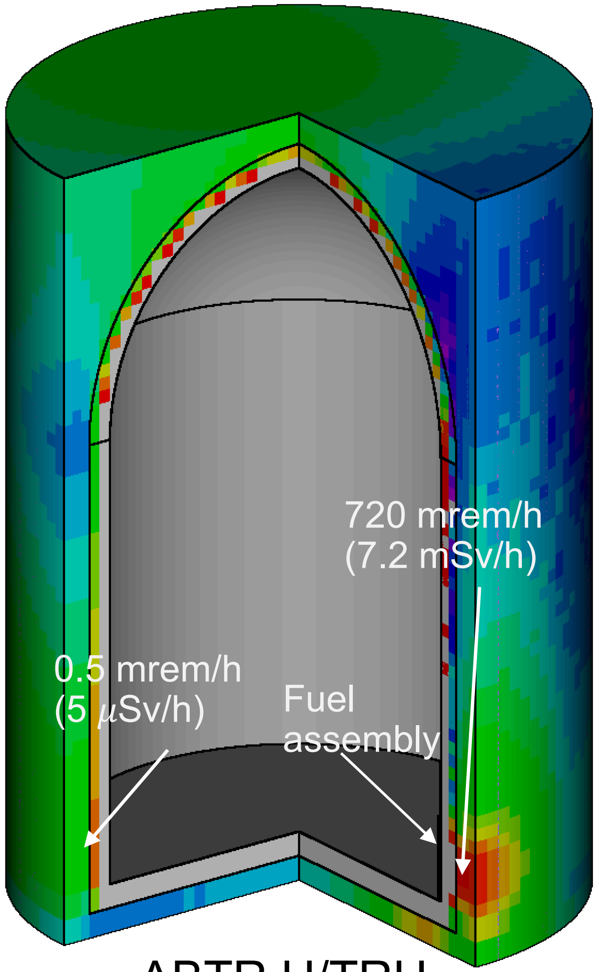


DOSE RATE MAPS OUTSIDE CB

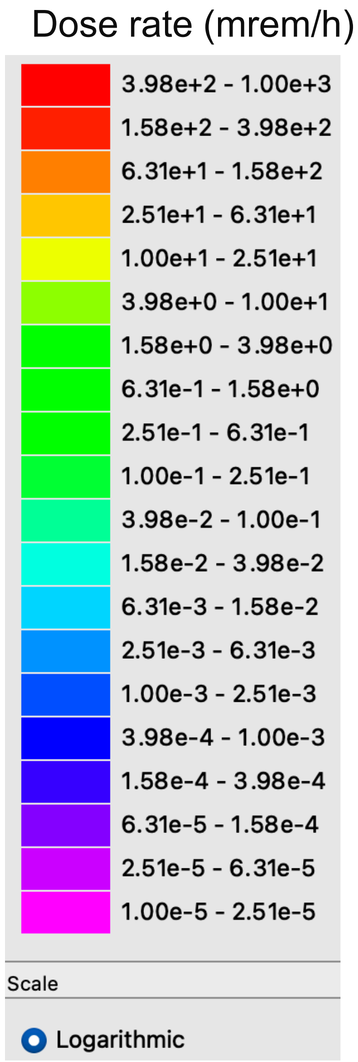
10 days cooling time



ABTR HALEU

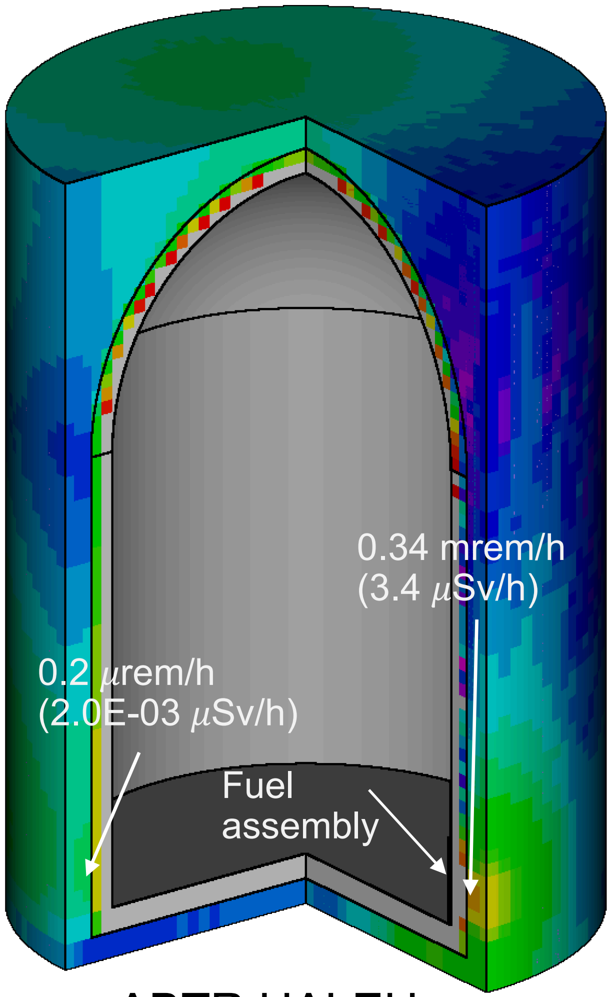


ABTR U/TRU

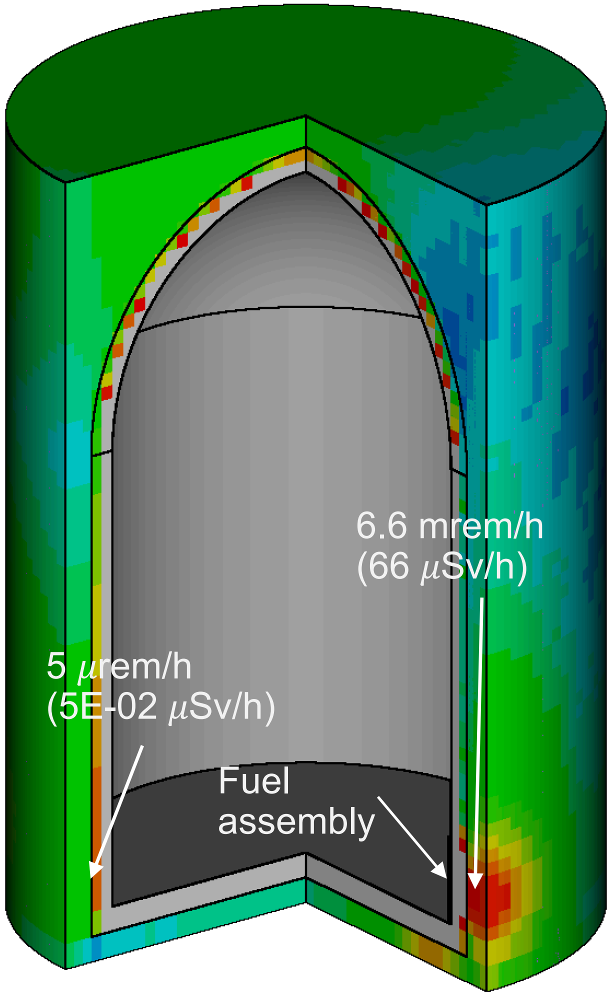


DOSE RATE MAPS OUTSIDE CB

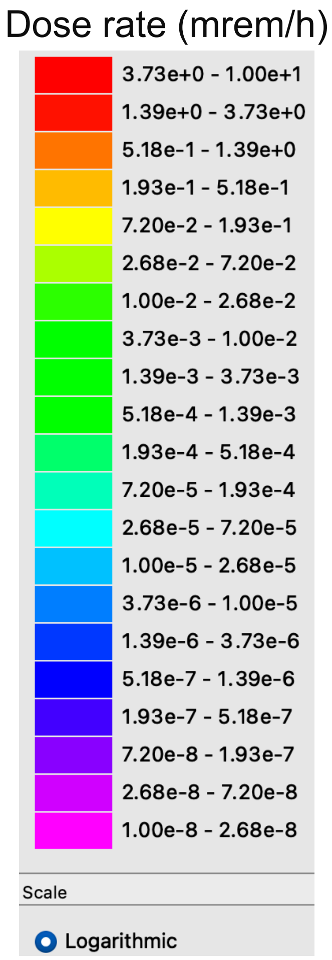
7 cycles of cooling time



ABTR HALEU



ABTR U/TRU



COMPARISON OF MAXIMUM DOSE RATES

Cooling time	Inside CB		Outside CB	
	ABTR HALEU	ABTR U/TRU	ABTR HALEU	ABTR U/TRU
10 days	4.6×10^{6} rem/h (4.6×10^{4} Sv/h)	6.0×10^{6} rem/h (6.0×10^{4} Sv/h)	530 mrem/h (5.3 mSv/h)	720 mrem/h (7.2 mSv/h)
7 cycles	9.2×10^{4} rem/h (9.2×10^{2} Sv/h)	1.9×10^{5} rem/h (1.9×10^{3} Sv/h)	0.34 mrem/h (3.4 μSv/h)	6.6 mrem/h (66 μSv/h)

- For comparison, the irradiation dose of PWR spent fuel (50 GWd/tHM) after 10 days of cooling is about 1.7×10^{6} rem/h (1.7×10^{4} Sv/h).
- Total dose rate dominated by primary gamma dose rate at these cooling times
- 10 CFR 20.1201 occupational annual dose limit for adults
  - Total effective dose equivalent (TEDE)* of 5 rems (0.05 Sv)

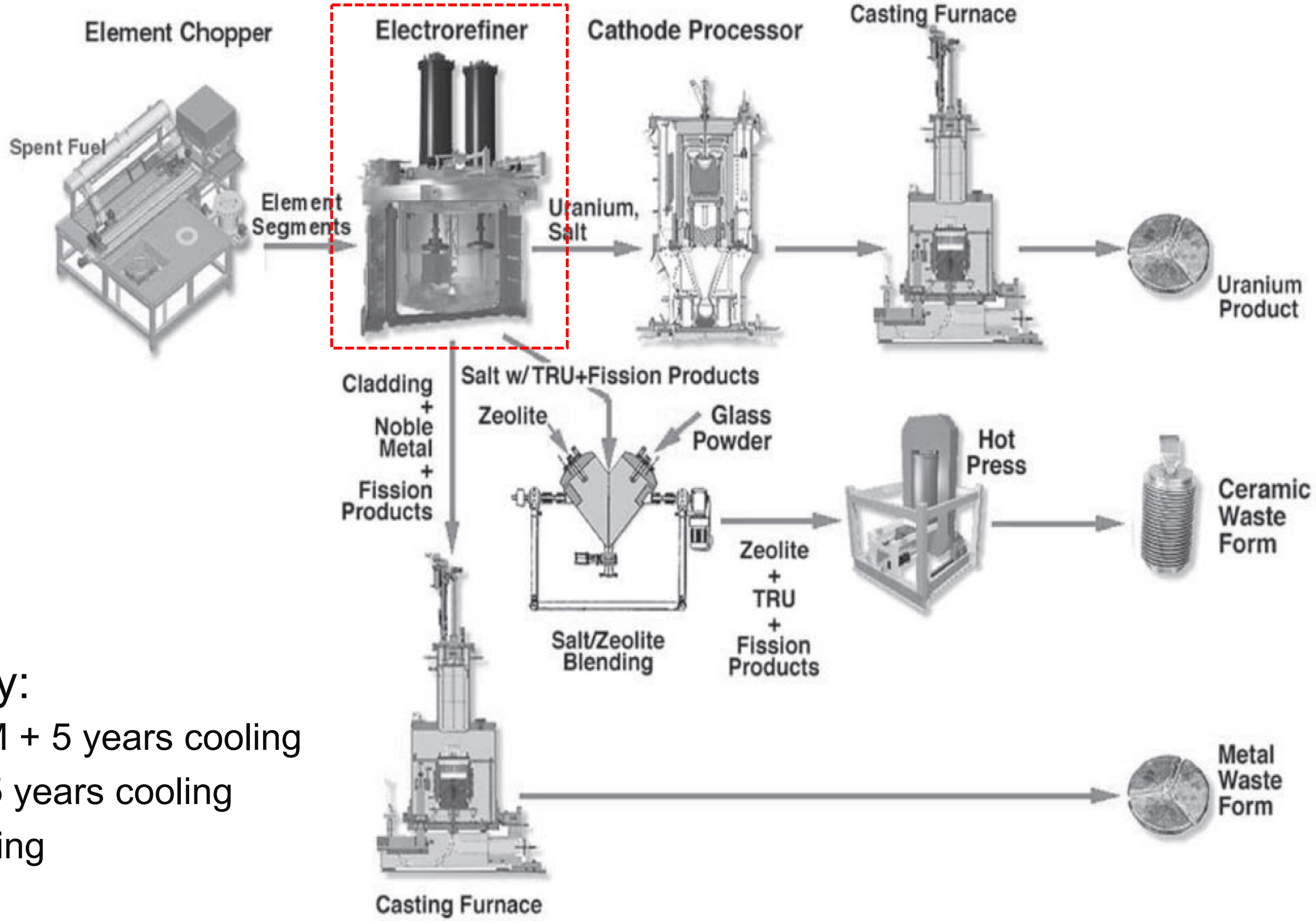
*TEDE means the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent (for internal exposures) (10 CFR 20.1003).

Scenario 2

Misfeed of material into the electro-processing batch leading to fissile material buildup / criticality as materials collect on the cathode

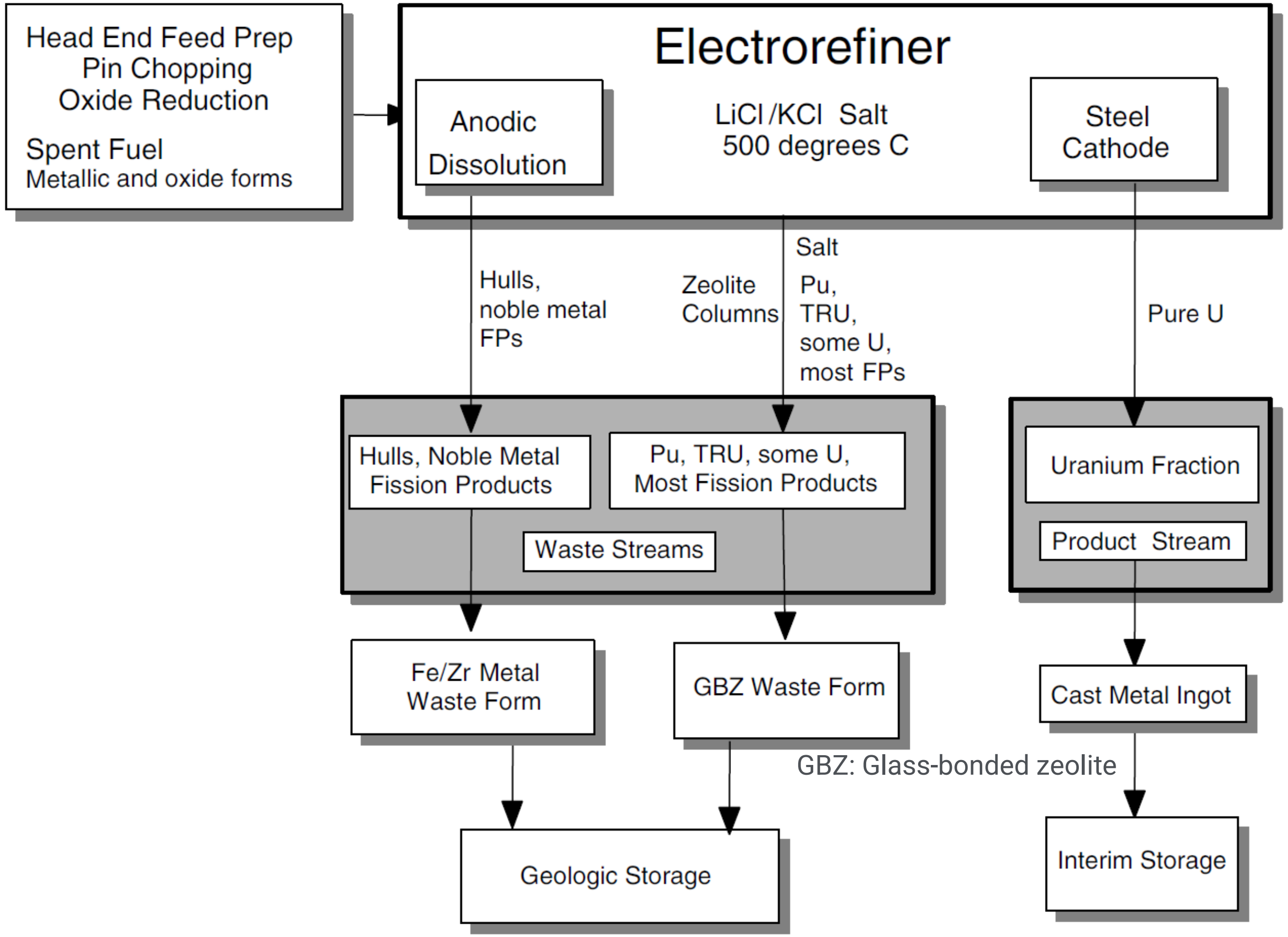
ELECTROMETALLURGICAL PROCESSING

- Electrometallurgical technology was originally proposed by ANL as a process to treat all DOE spent fuels.
- The analyses in this work were based on the experience for EBF II spent nuclear fuel treatment.
- The chopped PWR spent fuel wil undergo oxide reduction process (voloxidation) before electrorefining.
- Fuel assemblies irradiation history:
  - ABTR U/TRU (Inner): 94.5 GWd/tHM + 5 years cooling
  - ABTR HALEU: 149.74 GWd/tHM + 5 years cooling
  - PWR: 50 GWd/tHM + 10 years cooling



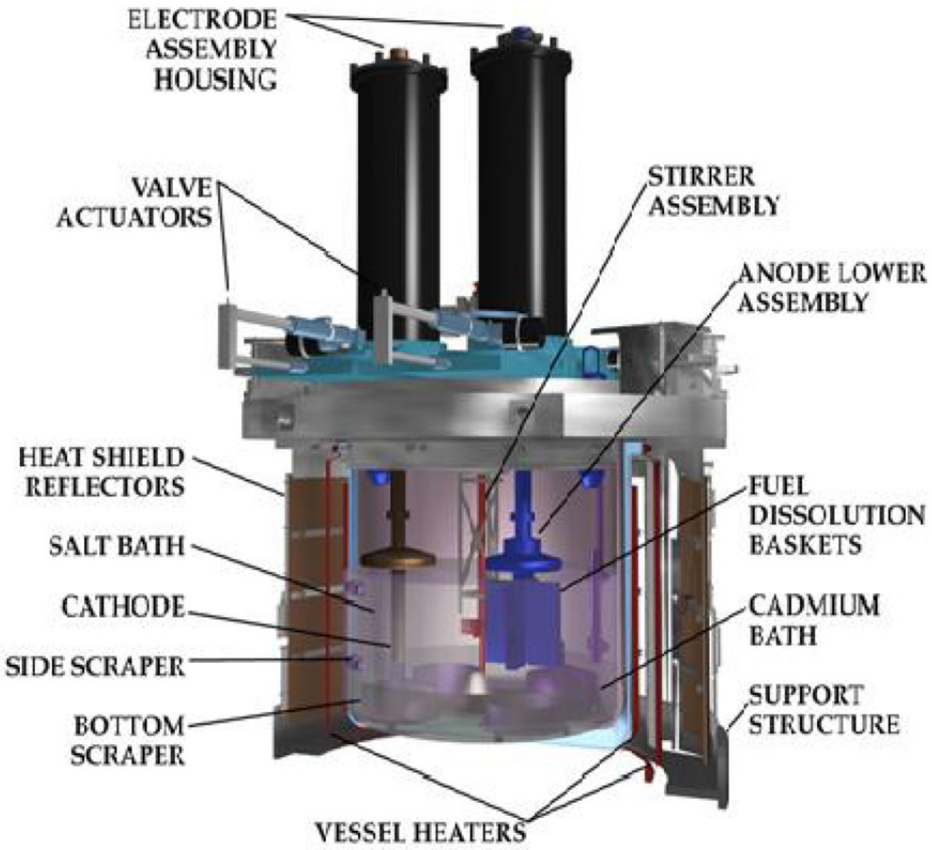
Ref.: National Research Council. 2000. Electrometallurgical Techniques for DOE Spent Fuel Treatment: Final Report. Washington, DC: The National Academies Press. https://doi.org/10.17226/9883.

ELECTROREFINING



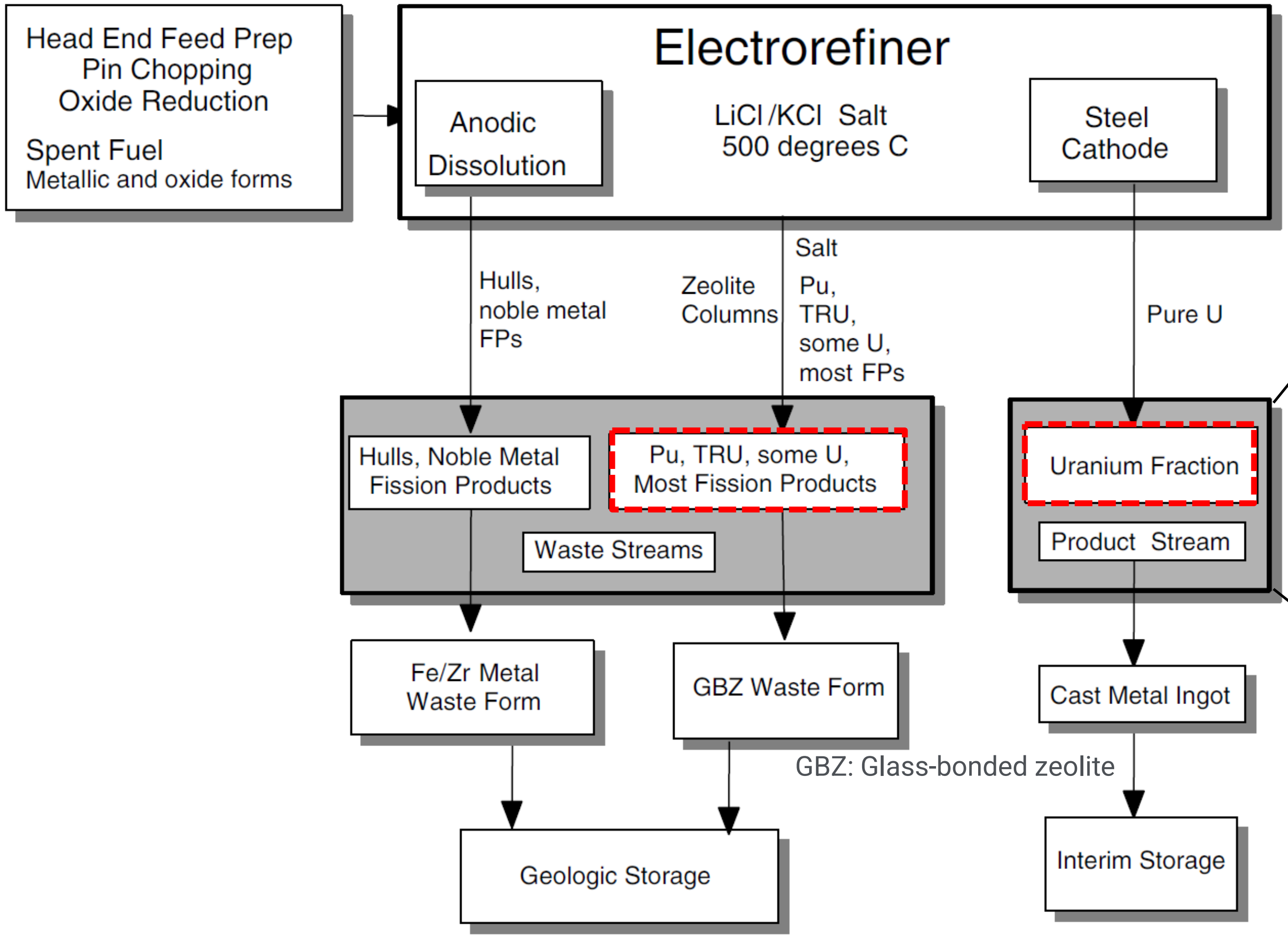
GBZ: Glass-bonded zeolite

Mark-IV Electrorefiner

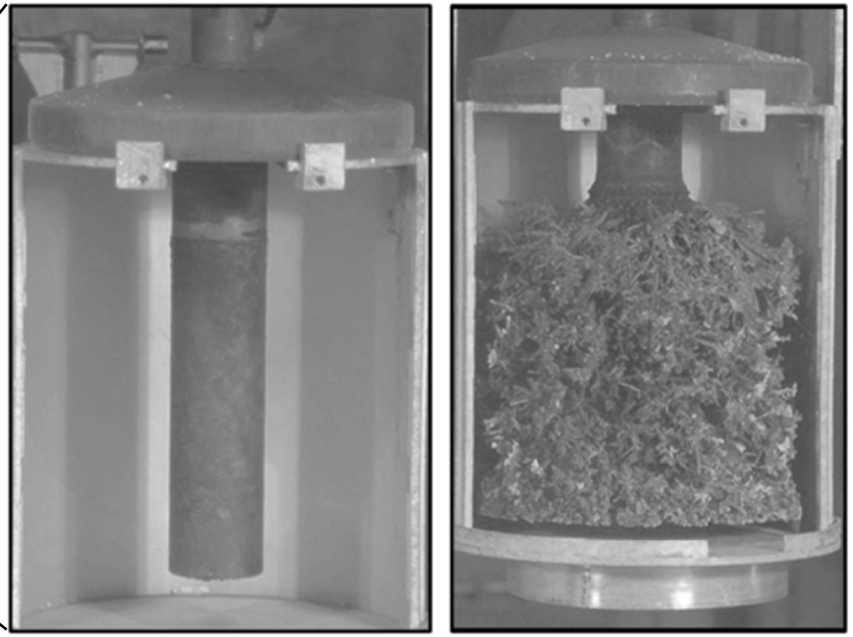


Ref.: National Research Council. 2000. Electrometallurgical Techniques for DOE Spent Fuel Treatment: Final Report. Washington, DC: The National Academies Press. https://doi.org/10.17226/9883.

ELECTROREFINING



GBZ: Glass-bonded zeolite

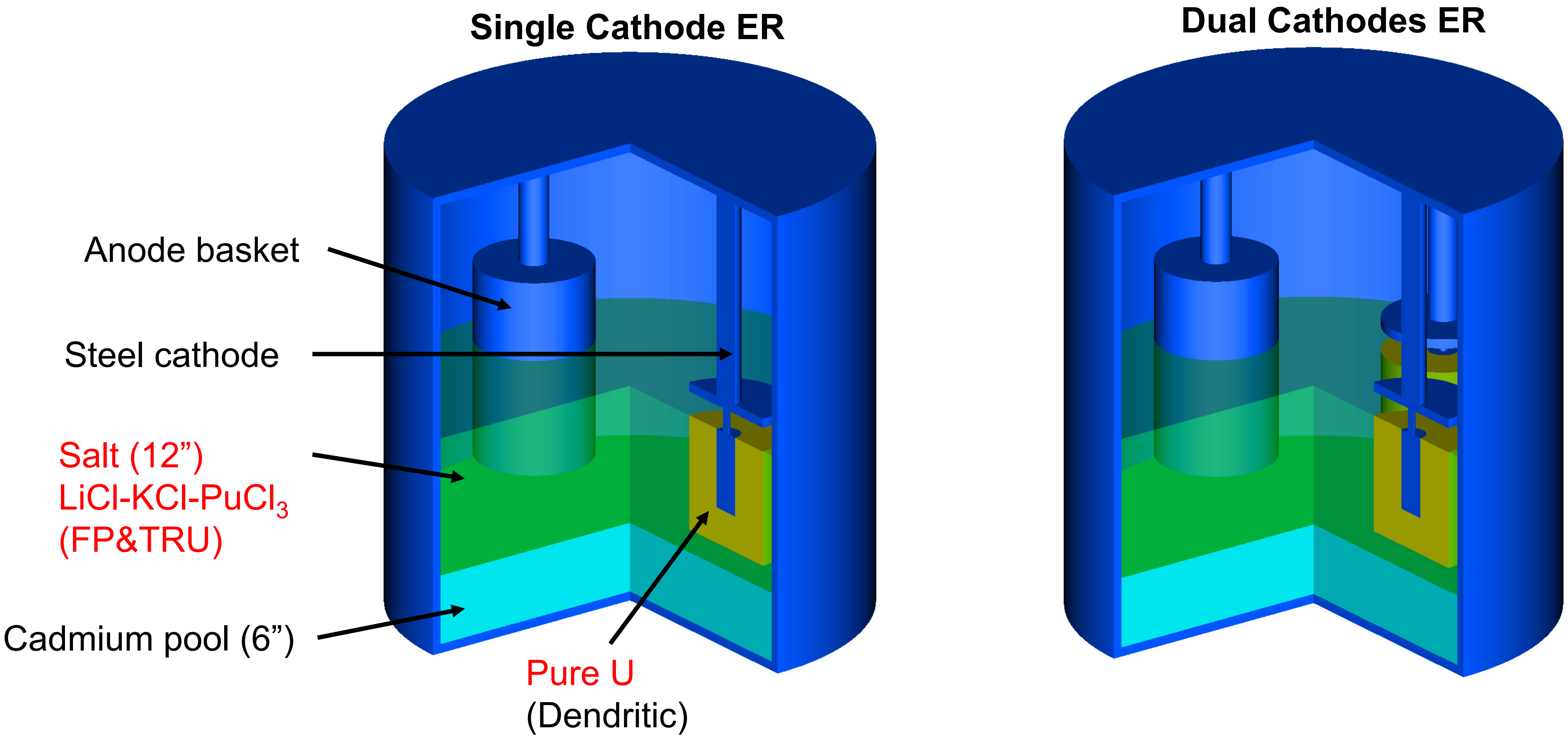


Ref: Fredrickson, G. L, et al. 2022. History and status of spent fuel treatment at the INL Fuel Conditioning Facility. Progress in Nuclear Energy 143, 104037, 2022.

Ref.: National Research Council. 2000. Electrometallurgical Techniques for DOE Spent Fuel Treatment: Final Report. Washington, DC: The National Academies Press. https://doi.org/10.17226/9883.

CRITICALITY ANALYSIS OF ELECTROREFINER

CSAS Model of Electrorefiner (40”x40”)

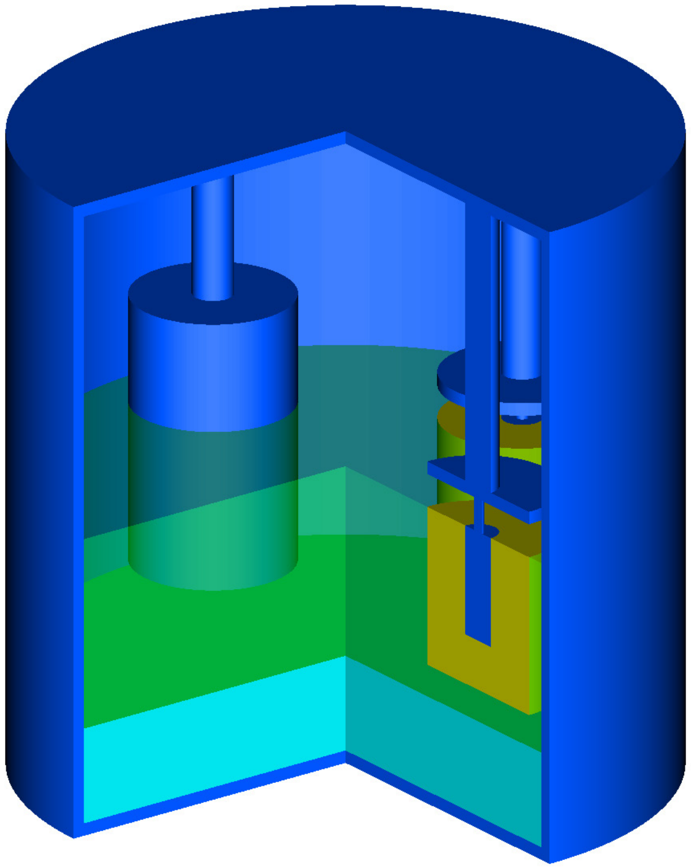


Ref.: *Robert D. Mariani, et al. Criticality Safety Strategy and Analysis Summary for the Fuel Cycle Facility Electrorefiner at Argonne National Laboratory West, Nuclear Technology, 114:2, 224-234, 1996.*

CRITICALITY ANALYSIS OF ELECTROREFINER

Vector of U and Pu in the recycled nuclear fuel

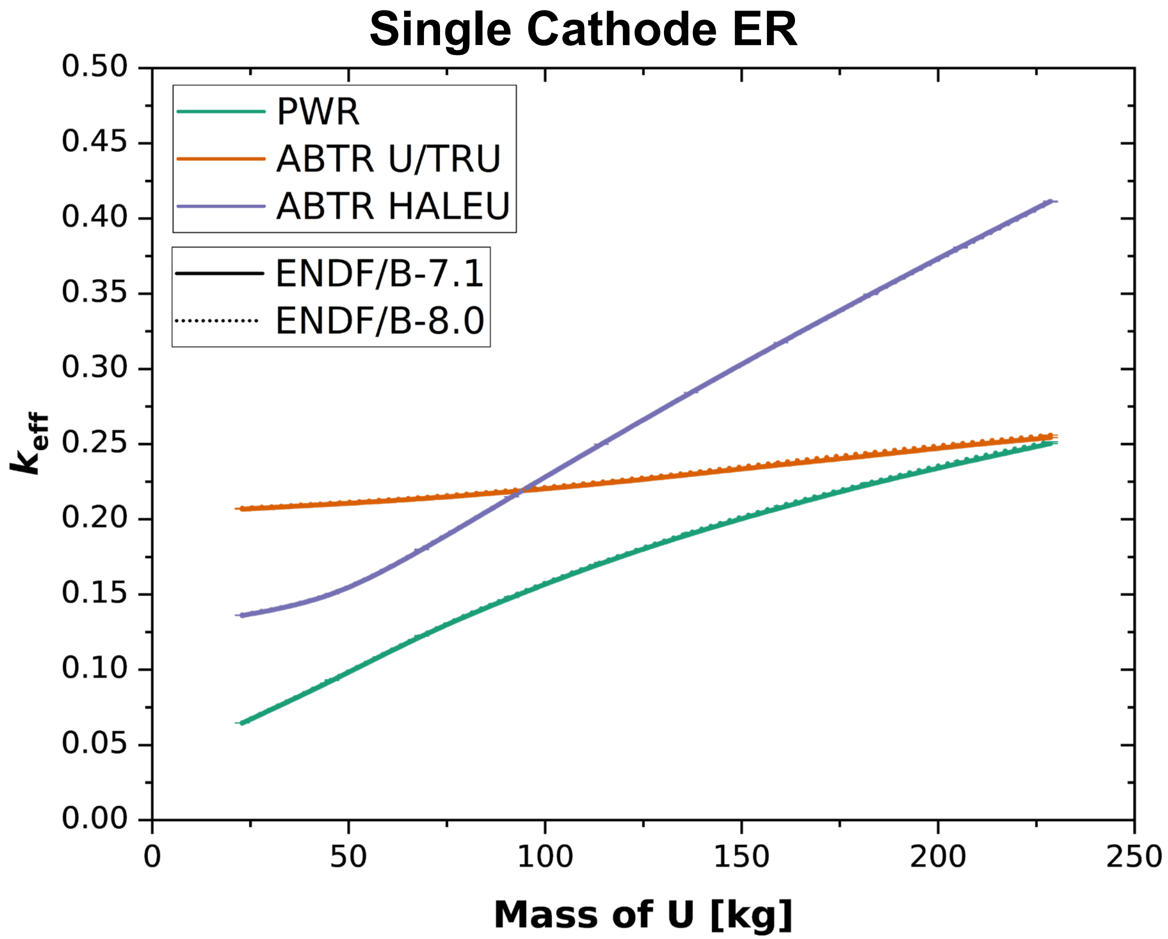
Fuel	U Vector	Pu Vector
U/TRU	0.01% U-234 **0.08% U-235** 0.03% U-236 99.88% U-238	0.53% Pu-238 **78.45% Pu-239** 18.81% Pu-240 **1.47% Pu-241** 0.74% Pu-242
HALEU	0.01% U-234 **5.81% U-235** 2.66% U-236 91.52% U-238	0.93% Pu-238 **88.83% Pu-239** 9.74% Pu-240 **0.47% Pu-241** 0.04% Pu-242
PWR	0.02% U-234 **0.80% U-235** 0.63% U-236 98.55% U-238	3.12% Pu-238 **54.18% Pu-239** 25.74% Pu-240 **9.59% Pu-241** 7.73% Pu-242

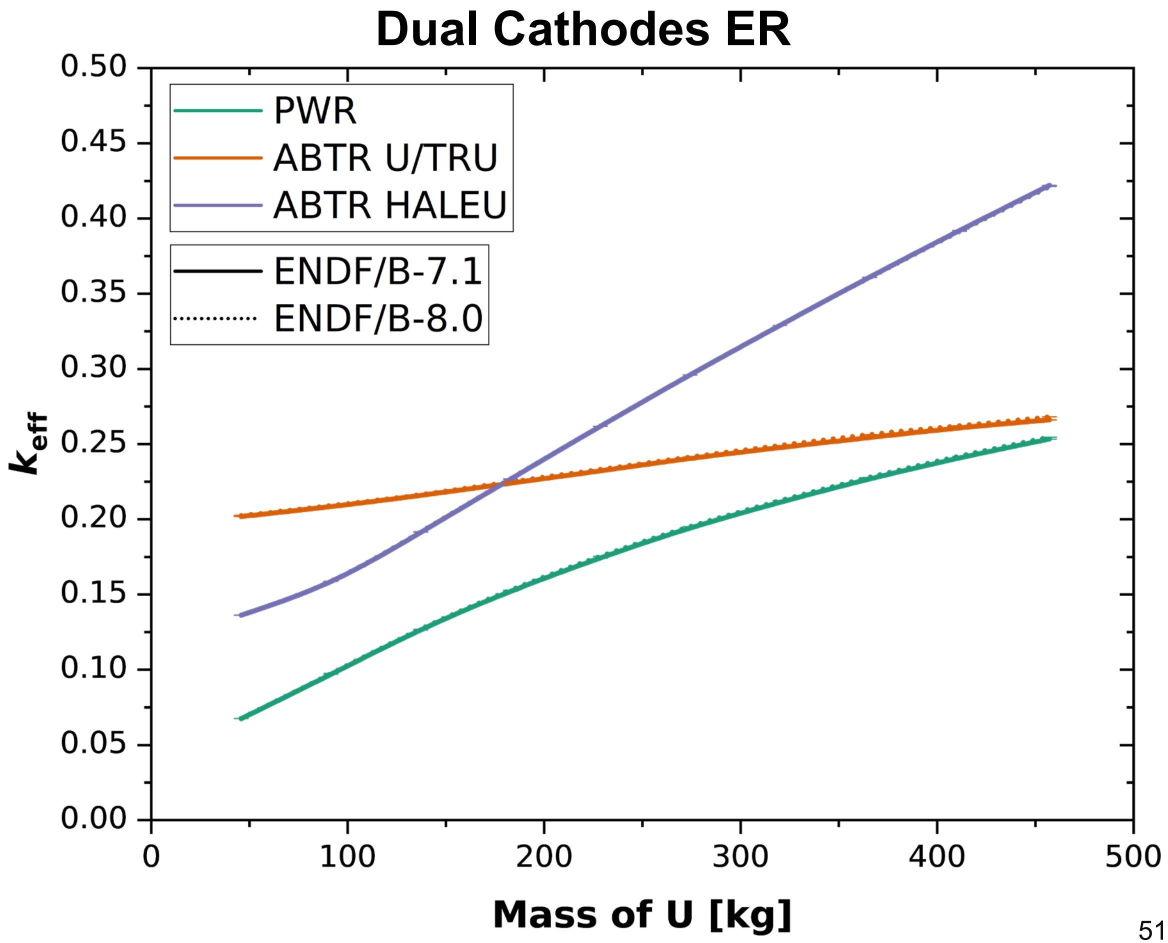


CRITICALITY ANALYSIS OF ELECTROREFINER

Multiplication factor as function of U mass in cathode

- k_{eff} is clearly below 0.95 even with the maximum U mass in the cathode
- Similar results were obtained by both nuclear data libraries (ENDF/B-7.1 and ENDB/B-8.0)

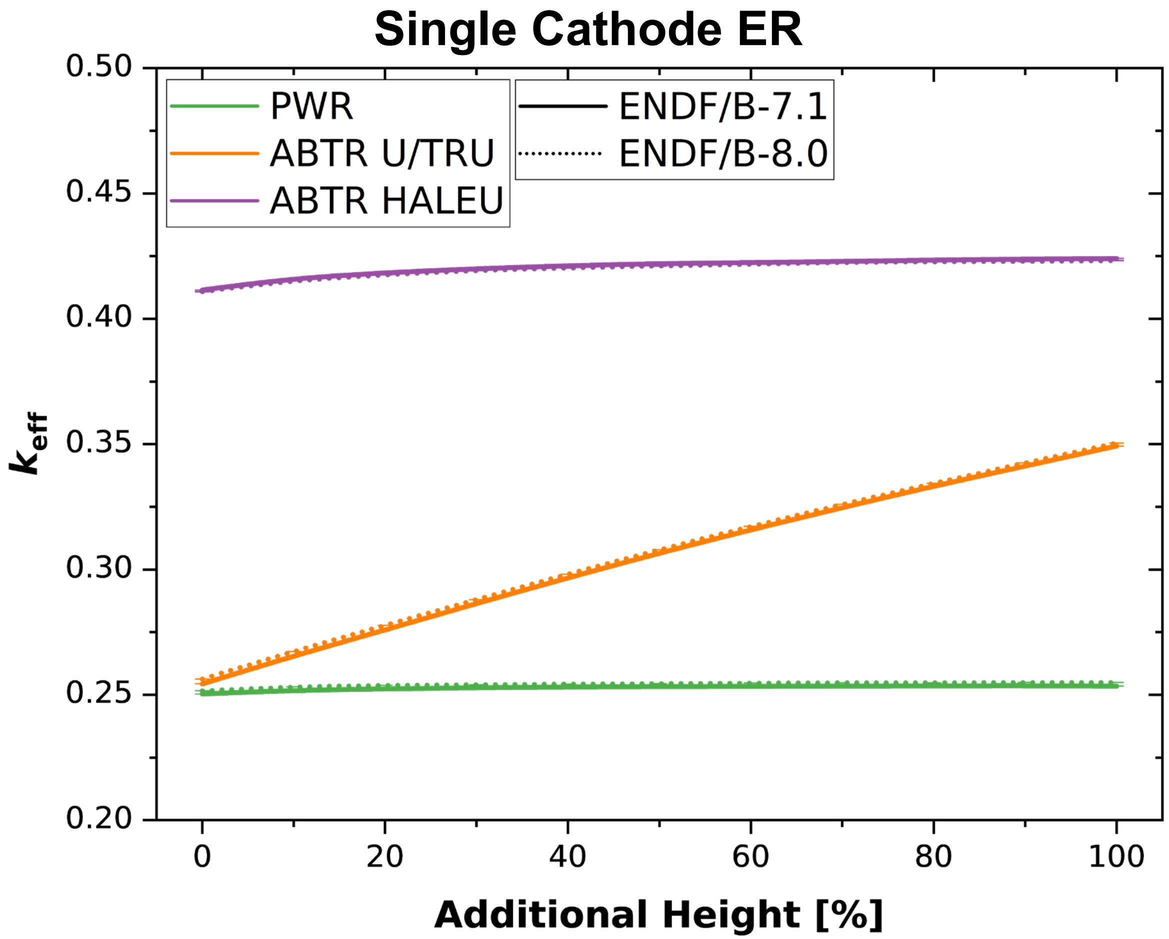


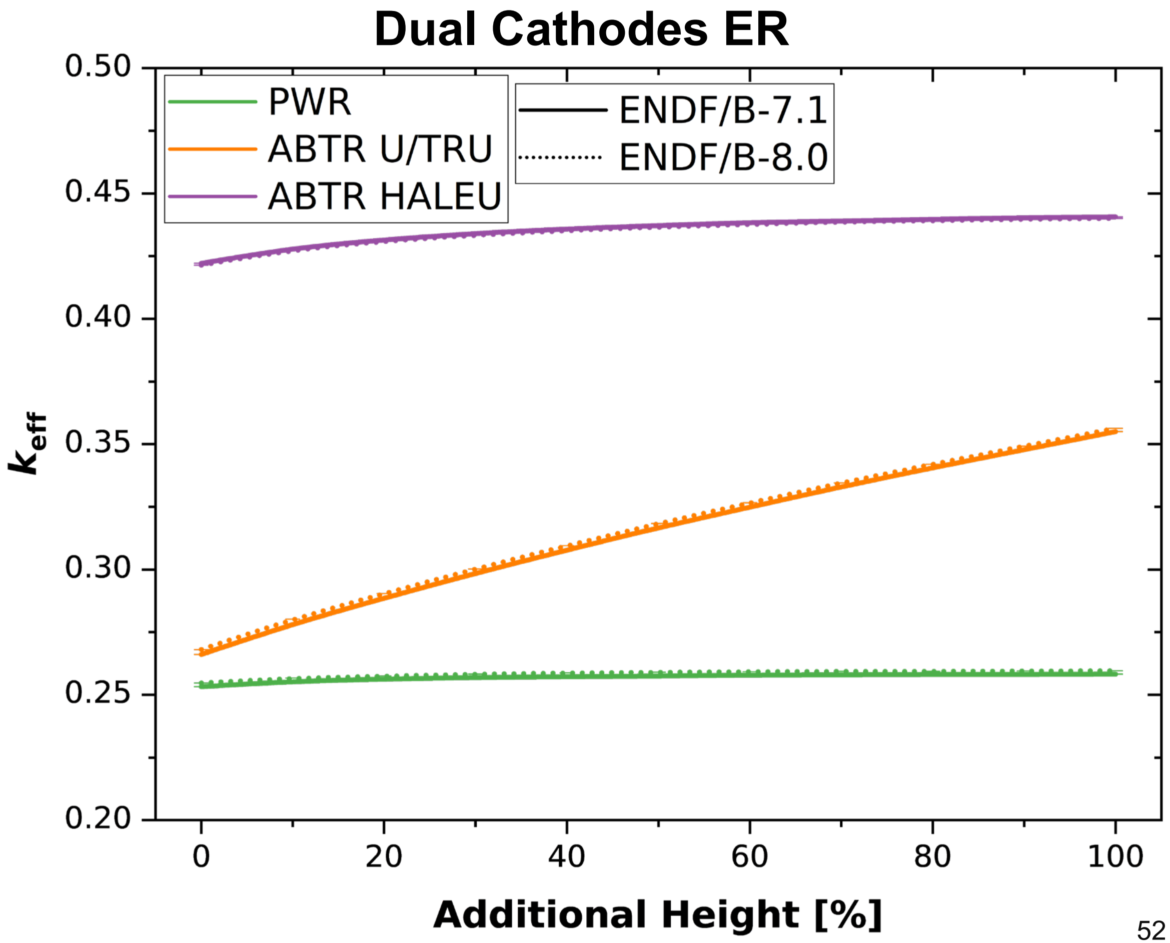


CRITICALITY ANALYSIS OF ELECTROREFINER

Multiplication factor as function of salt height in the tank

- k_{eff} does not change significantly when the salt height increases, and remains clearly below 0.95



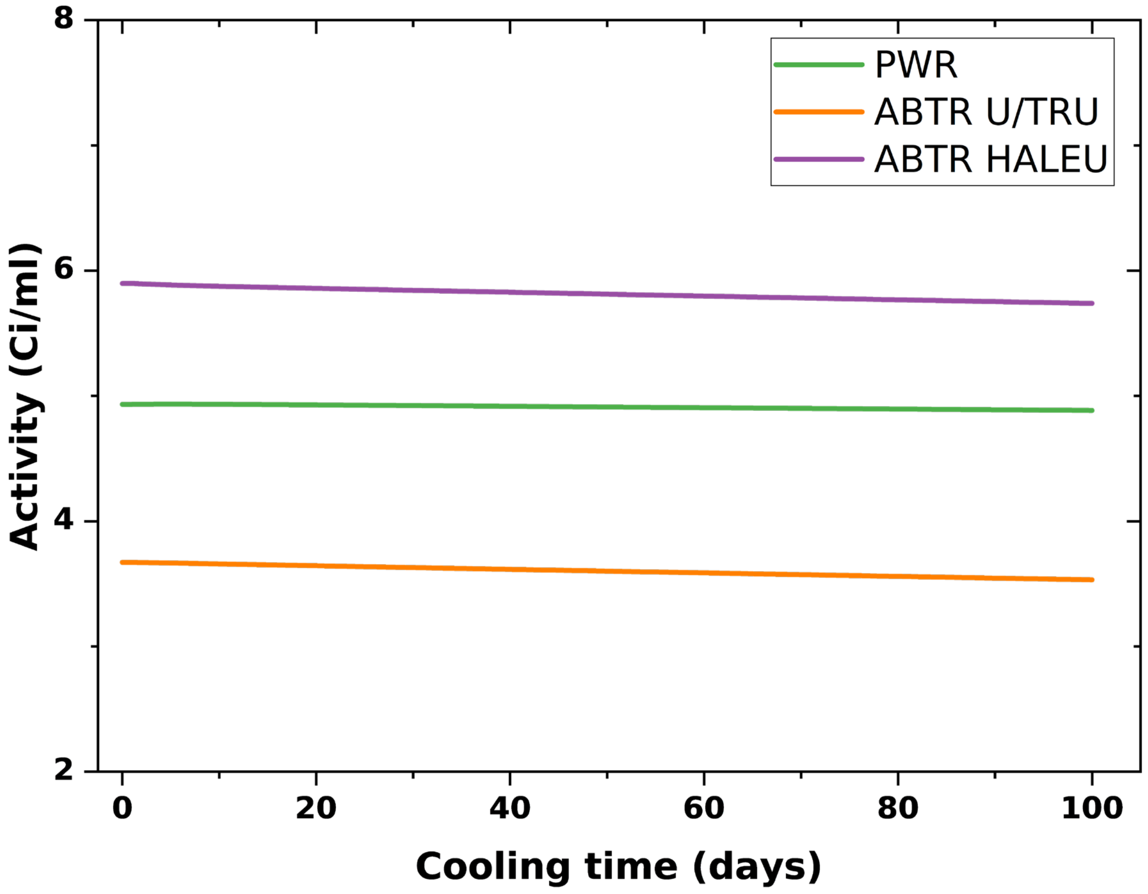


Scenario 3

A leak in the waste stream storage tank allows for release of fission products during reprocessing

ACTIVITY OF SALT

- According to IAEA Technical Reports Series No. 135 (1972), an activity > 10^{-2} Ci/ml requires cooling and shielding.
- Currently salt is assumed to contain 10 wt.% $PuCl_3$. Pu in $PuCl_3$ lumps all TRU and majority of the fission products.



Summary

Summary & Outlook

- SCALE capabilities to simulate different scenarios in the different SFR fuel cycle stages were demonstrated.
- The demonstrated capabilities included the rapid calculation of fuel inventory, decay heat and activity, as well as shielding, radiation dose, and criticality calculations.
- Key observations:
  - The radiation dose of the ABTR spent fuel is significant, requiring proper shielded when removed from the core
  - ABTR fuel assembly dose rate is dominated by the fuel’s gamma sources
  - Criticality analyses of electrorefiner show k_{eff} << 0.95 in all considered configurations
  - Shielding and cooling may be required for the liquid waste salt from electrorefiner

Summary & Outlook

- Additional information is needed for improved analysis:
  - Detailed information on the salt mixtures during reprocessing
  - Onsite storage of fresh and irradiated fuel assemblies (storage containers, storage configuration, etc.)
  - Commercial size transportation canisters for UF_6, reprocessed PWR and SFR fuel, spent SFR fuel
- Related future development in SCALE:
  - Development of a strategy for SFR equilibrium core generation
  - Efficient reactivity feedback calculation
  - Integration of simple thermal expansion model
- Reference SCALE ABTR 3D models are available online
  - Repository: https://code.ornl.gov/scale/analysis/non-lwr-models-vol3
  - Vol. 5 models will be added soon

SAND2023-08740PE

Demonstration of MELCOR for SFR Fuel Cycle Analysis

KC Wagner, David L. Luxat

MELCOR Application to Fuel Cycle Safety Assessment

MELCOR is used in the DOE complex for facility safety analysis

MELCOR has **general and validated** models for thermal hydraulic behavior of enclosures and hazardous material transport

- Enables modeling of potential for fission products to be released from an enclosure to the environment

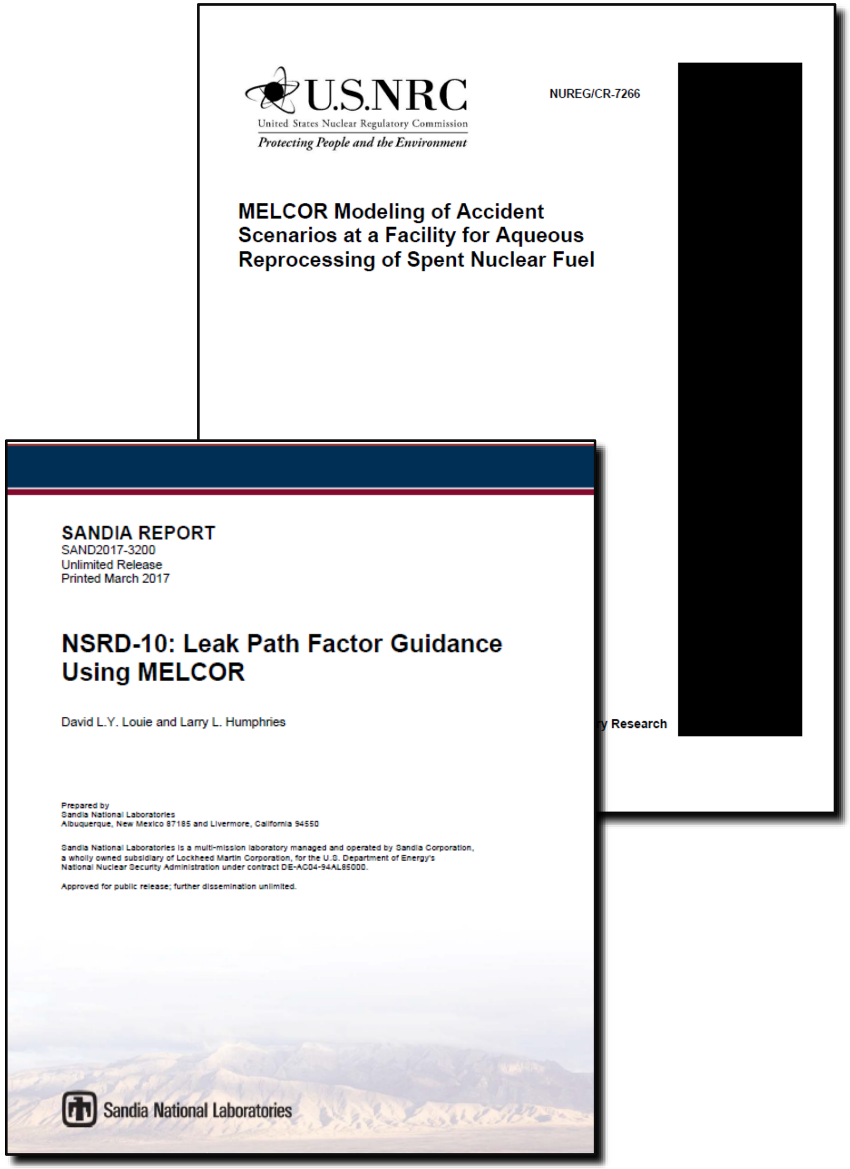
MELCOR has been applied to safety basis development for a broad range of facility accidents that can lead to accident release of hazardous material

- Inadvertent nuclear criticality events
- Explosions
- Broad range of facility fires
- Radioactive material spills and drops

MELCOR enables assessment of a range of conditions that can impact hazardous material release to the environment

- External winds promoting enhanced transport from an enclosure to environment
- Retention of hazardous material in filters
- Removal of hazardous material from enclosure atmospheres by decontamination sprays

Recent NRC research application of MELCOR to demonstration of safety assessment at Barnwell reprocessing facility


U.S.NRC
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Protecting People and the Environment

NUREG/CR-7266

MELCOR Modeling of Accident Scenarios at a Facility for Aqueous Reprocessing of Spent Nuclear Fuel

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NSRD-10: Leak Path Factor Guidance Using MELCOR

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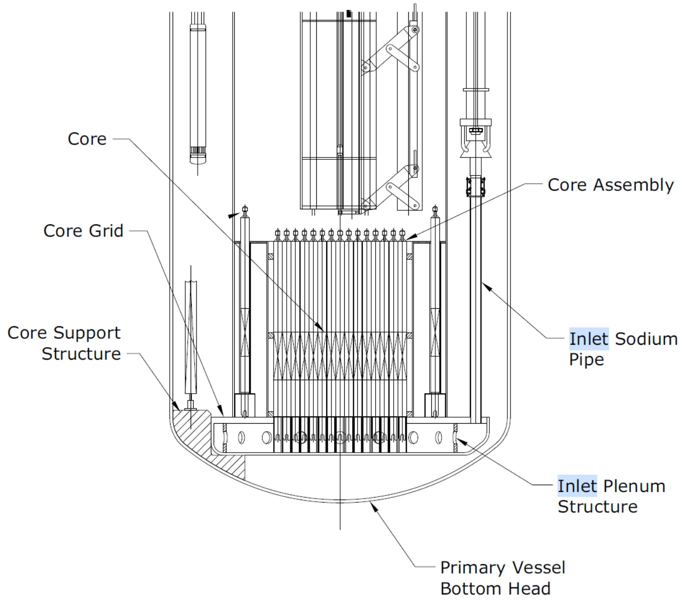
Modeling SFR Accidents with MELCOR

- **SFR materials**
  - U-10Zr metallic fuel, HT-9 cladding, and sodium bond
  - Sodium fluid EOS
- **SFR Fuel Representation**
  - 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
- **Reactivity accidents**
  - Reactor point kinetics and application to fast reactors
- **SFR Fuel Degradation**
  - Clad pressure boundary failure, melting and candling
  - Fuel melting
  - Degraded fuel region molten and particulate debris behavior
- **Radionuclide release and transport**
  - Gap and plenum release
  - Molten fuel fission gas release
  - Thermal release models
- **Sodium pool and spray fire models**



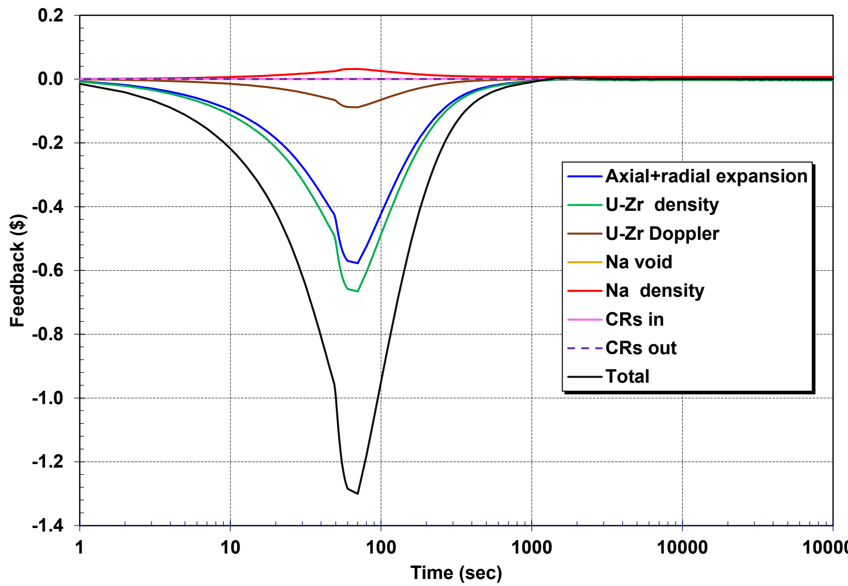
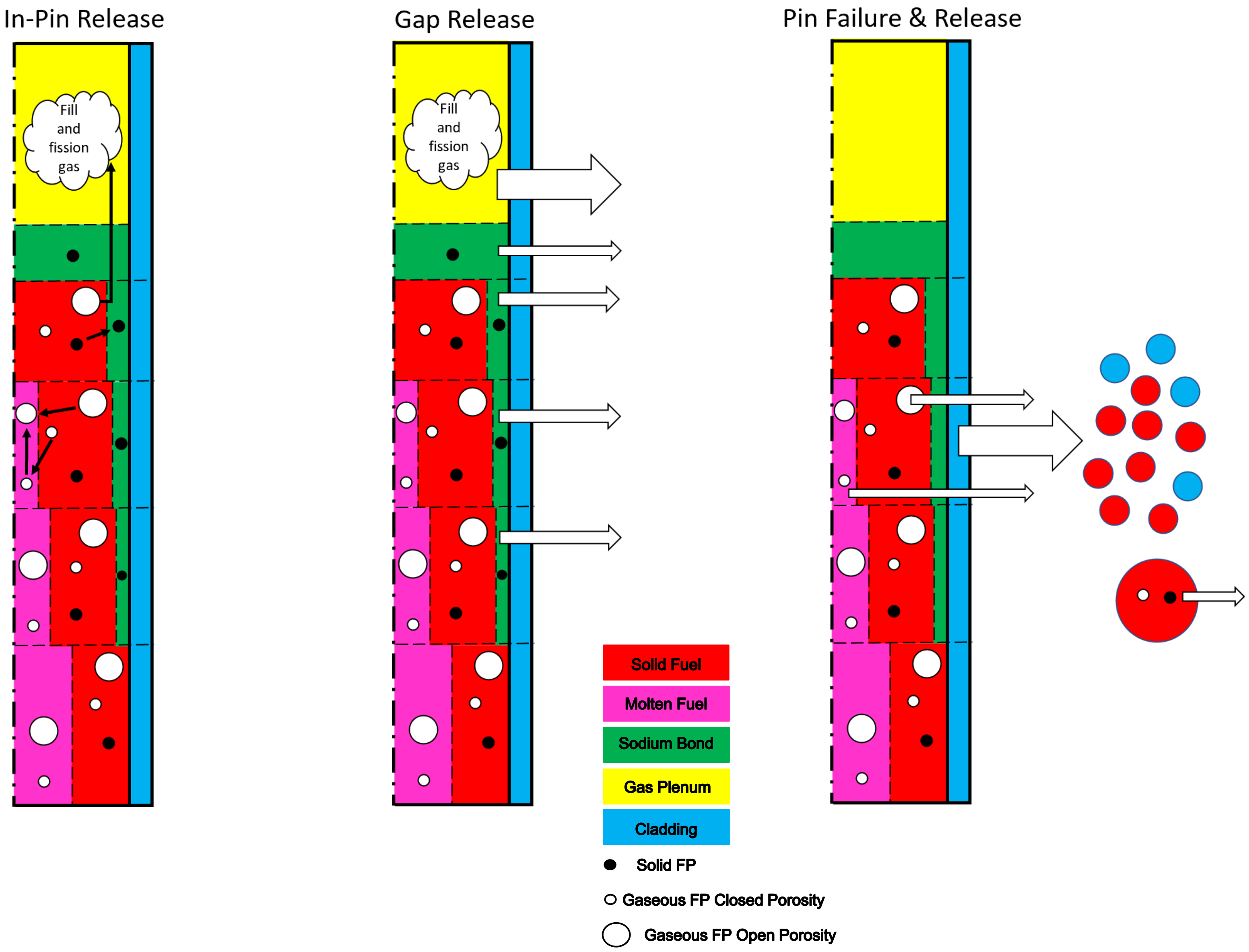


Figure Ref. Y.I. Chang, P.J. Finck, and C. Grandy, “Advanced Burner Test Reactor Preconceptual Design Report”, ANL-ABR-1 (ANL-AFCI-173), 2006.]

Capability: Fission Product Release from SFR Fuel

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



Capability: Fission Product Release from Sodium Coolant

Goal: *Determine magnitude of fission product release into enclosure atmospheres and available to release to environment*

Fission product release into sodium coolant from fuel upon cladding failure

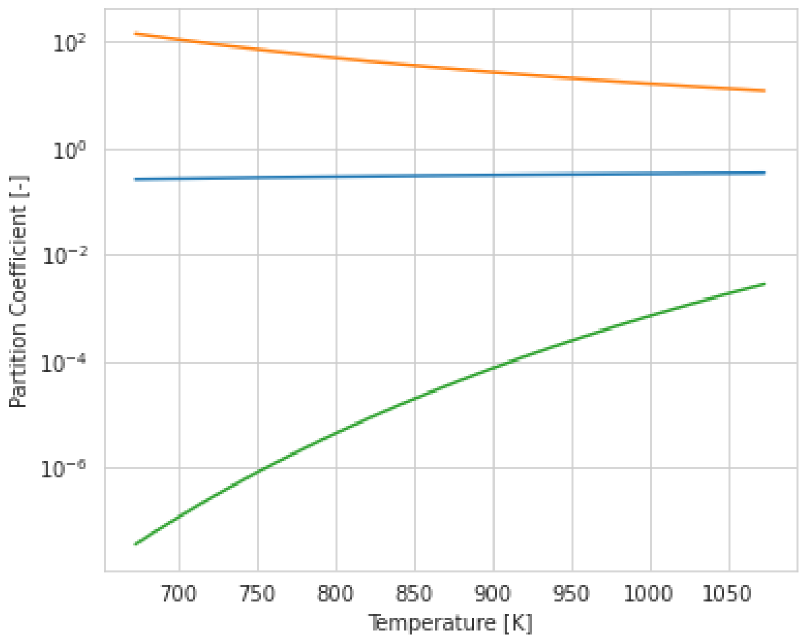
- What fraction of fission products in the sodium are available to be released from sodium?
- Chemical interaction of fission products with sodium critical to determine volatility of fission products

Distribution of fission products in sodium influences transport out of sodium

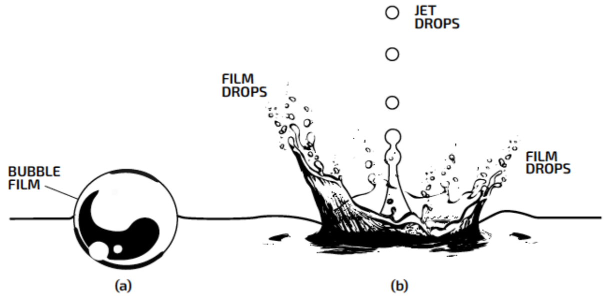
- Dissolved in sodium
- Colloidal particles in sodium
- Gaseous in sodium
- Deposited on structures interfacing with sodium

Transport paths out of working fluid like sodium being considered in development

- Evaporation influenced by solubility and vapor pressure
- Bubble transport and bursting
- Mechanical mobilization through jet breakup and splashing



Haga et. al., Nuclear Technology **97**, 177 (1992)



Capability: Sodium Fire Modeling and Impact on Fission Product Mobilization and Transport

Sodium reacts with oxygen and water

(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)

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

Atmospheric chemistry + aerosol generation

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

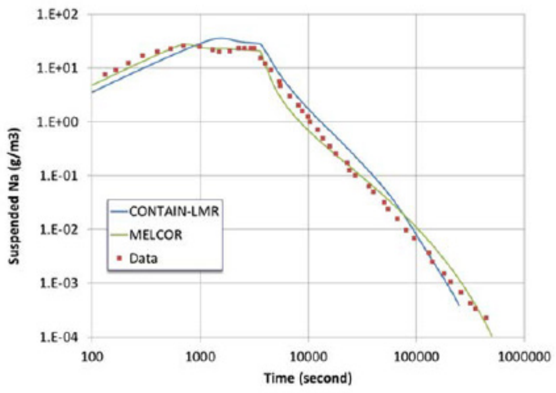
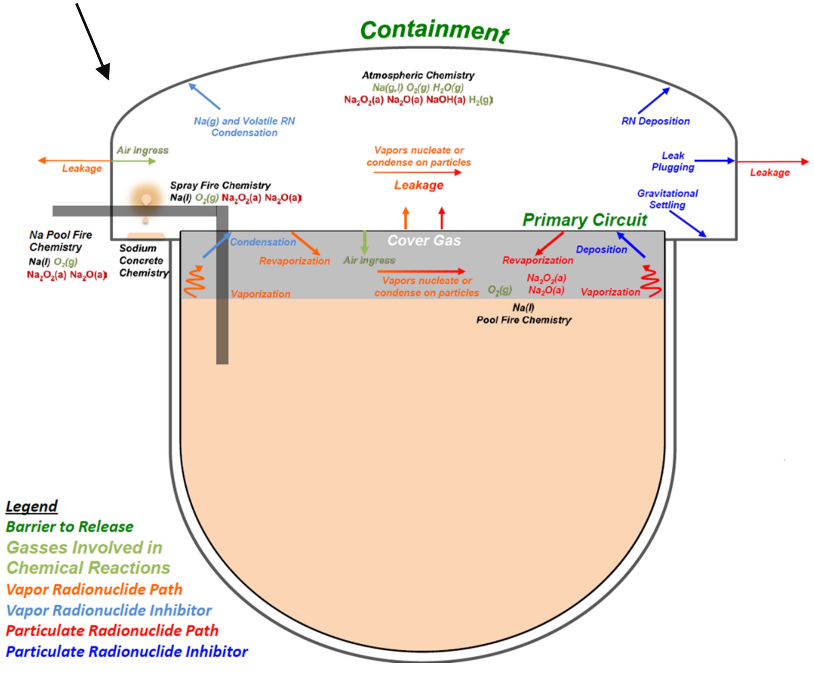


Figure 33. Suspended Na Aerosol Mass - AB1



[Figure adapted from ANL-ART-3]

Containment and Reactor Building

ABTR defense in depth features included in the MELCOR modeling –

- Primary containment boundary
  - Reactor vessel
  - Reactor vessel enclosure (top closure of the vessel with refueling port)
  - Intermediate heat exchanger tubes
  - Direct Reactor Auxiliary Cooling System (DRACS) heat exchanger tubes
  - Sodium purification piping and components

- Secondary reactor building boundary
  - Reactor guard vessel (nitrogen-inerted)
  - Reactor containment dome
  - Sodium-to-CO_2 heat exchangers
  - DRACS intermediate system piping and systems
  - Stainless steel-lined compartments around the vessel
  - Purification system cell confinement
  - Reactor building

Containment and Reactor Building

Key sodium support systems

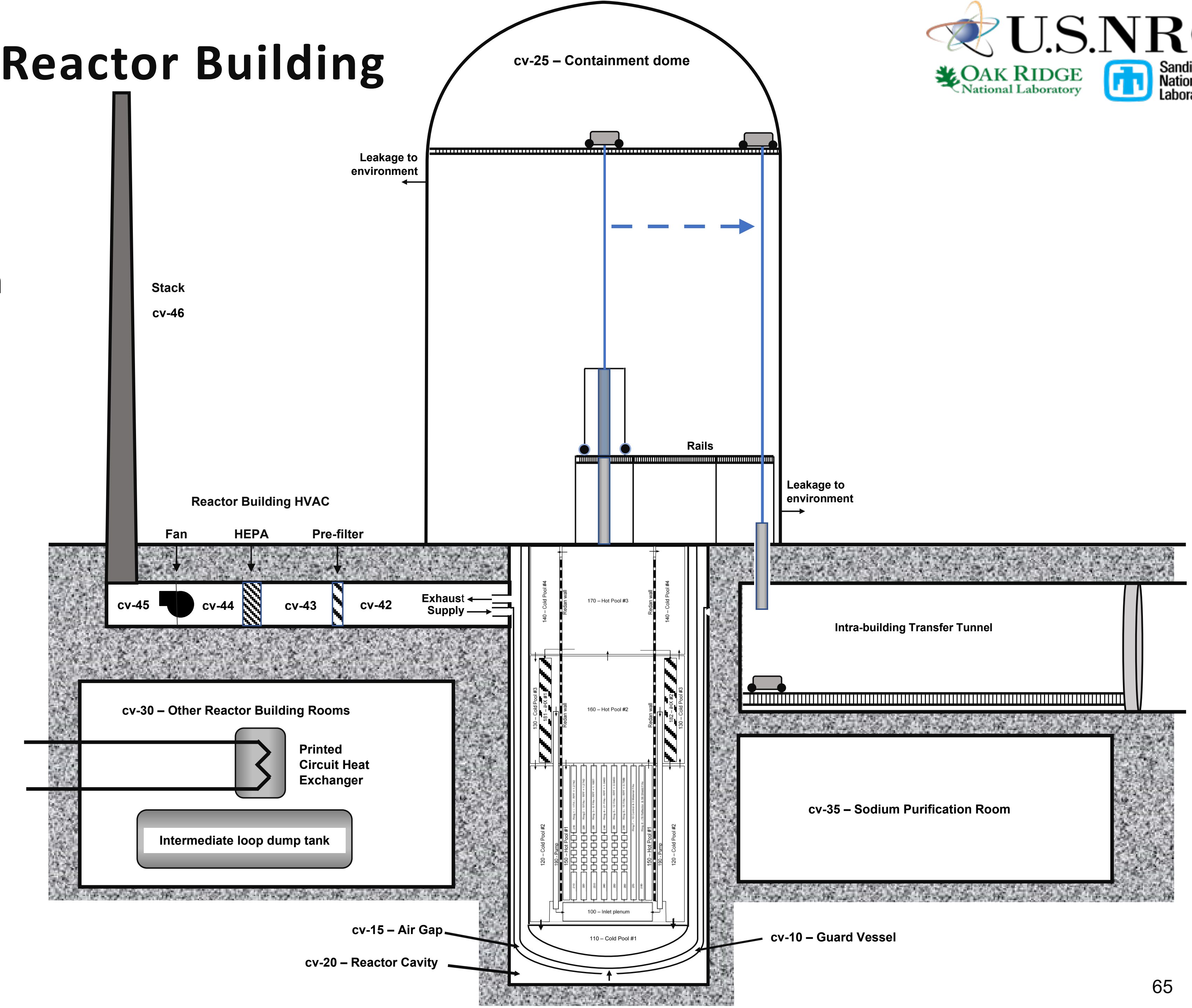
- Sodium purification system
- Argon cover gas purification system

ABTR design leak rate is consistent with LWR containments

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

HEPA-filtered ventilation system

- 2X air exchanges per hour (assumed)
- Maintains -2” H_2O reactor building pressure



Scenarios

Fuel Unloading Machine (FUM) failure scenario

- Cask drop with leak in the containment dome

Sodium purification pipe break during operations with coincident fuel clad failure and activity release

- Use integrated primary system core damage models with equivalent of 217 fuel rod clad failures (i.e., 1 assembly)
- Sodium fire in the Sodium Purification room

Argon cover gas piping failure with coincident fuel clad failure and activity release

- Use integrated primary system core damage models with equivalent of 1-assembly clad failures
- Contaminated argon discharges into the Sodium Purification room

Reprocessing accident scenario capability discussion

- Illustrations from Barnwell safety analysis for pyro-refining or fuel fabrication plants

Fuel Unloading Machine (FUM) failure scenario

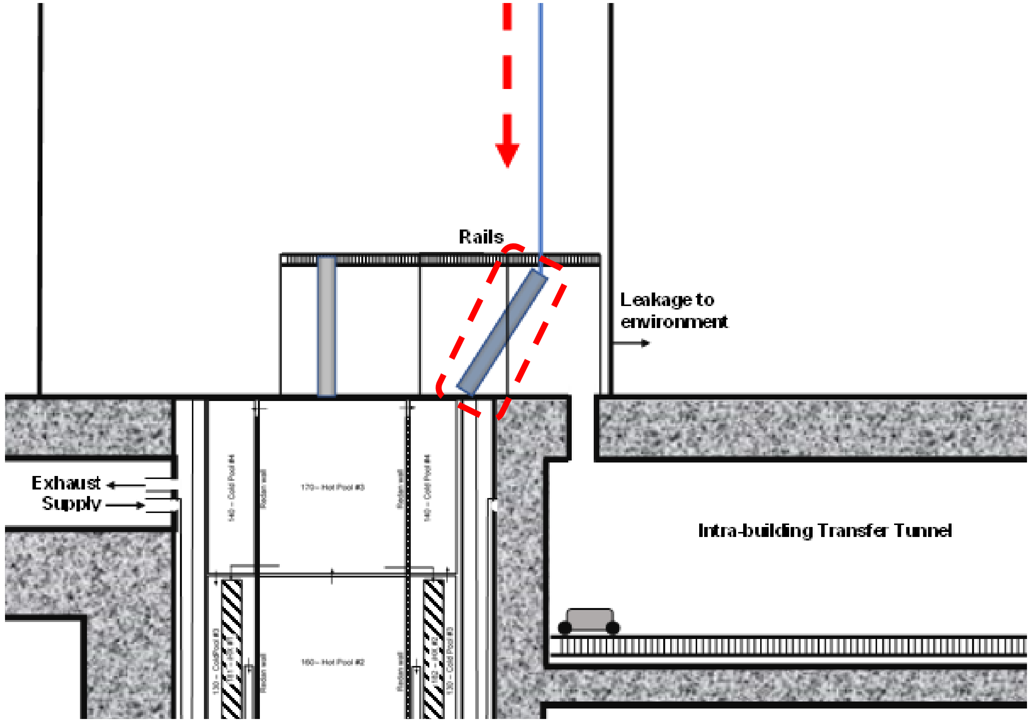
FUM is used to load, unload, and move fuel

- The FUM connects to the reactor enclosure for refueling operations
- The ABTR in-vessel fuel rack can hold 36 assemblies
- Recently discharged fuel is moved into racks for in-vessel storage (IVS)
- Fuel remains in IVS for ~7 fuel cycles (~28 months)
- FUM moves used fuel storage vault via the intra-building transfer tunnel

MELCOR fuel damage model used to represent in the FUM

SCALE provided fuel radionuclide inventories

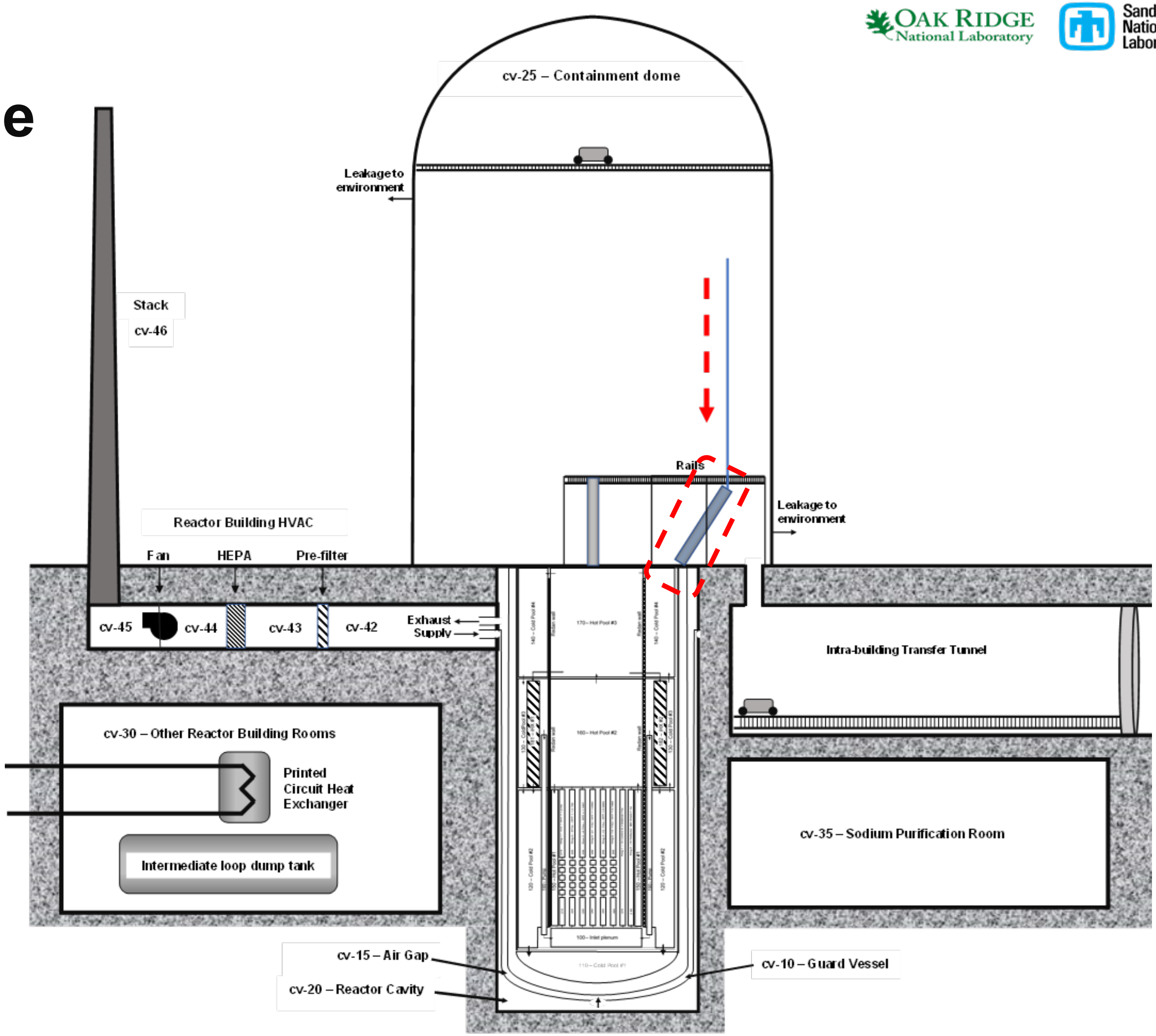
- HALEU spent fuel after in-vessel storage (IVS)
- Inner Transuranic (TRU) fuel after IVS
- Outer TRU fuel after IVS
- HALEU fuel after irradiation



Fuel Unloading Machine (FUM) failure scenario

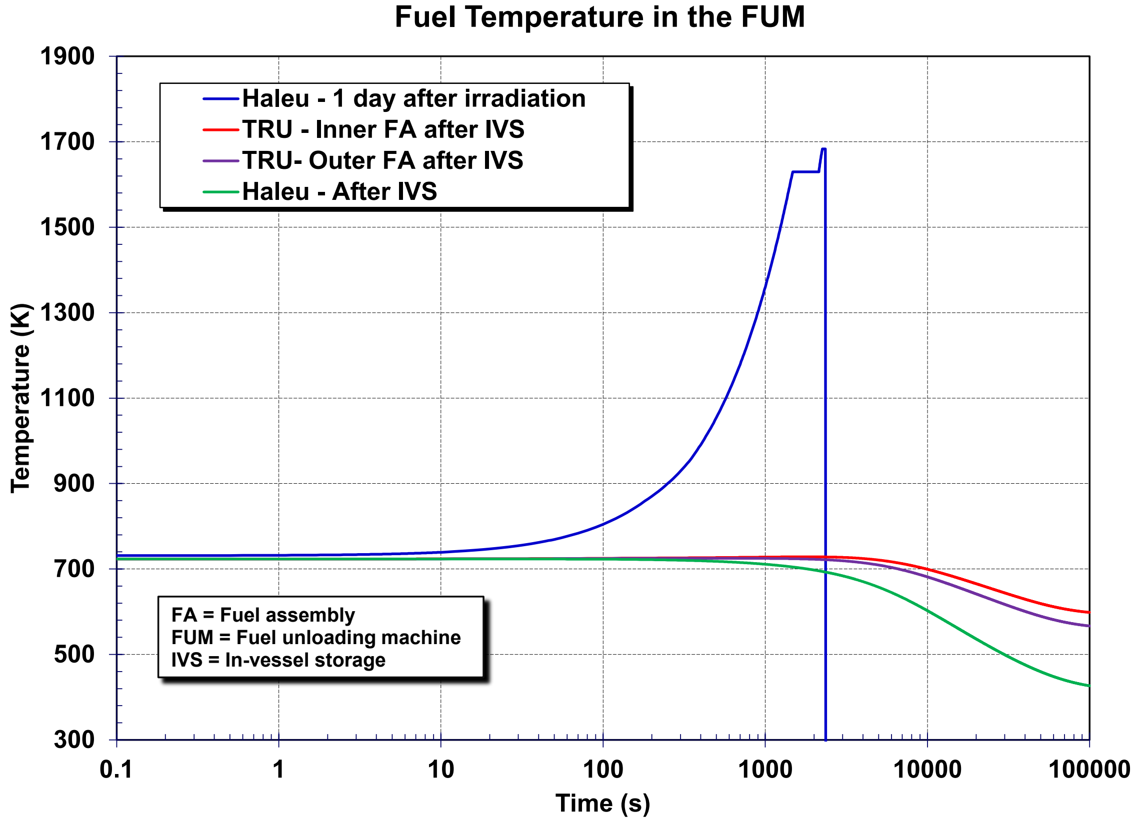
Accident scenario assumptions

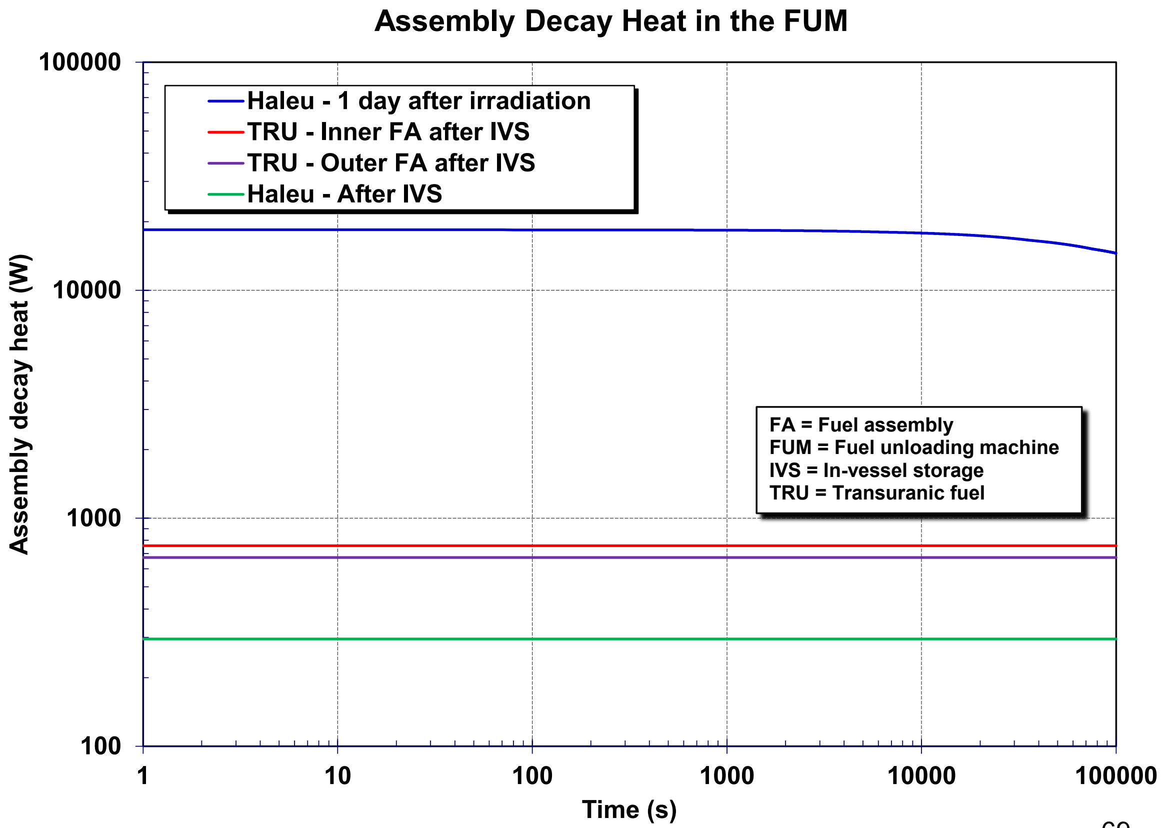
- High and low leaks in FUM cask
- Reactor building HVAC is filtering the containment dome during refueling operations
- No residual sodium in the cask
- All active cooling systems have failed
- Last case uses a fuel assembly accidentally removed with only 1-day cooling after last irradiation



FUM accident scenario

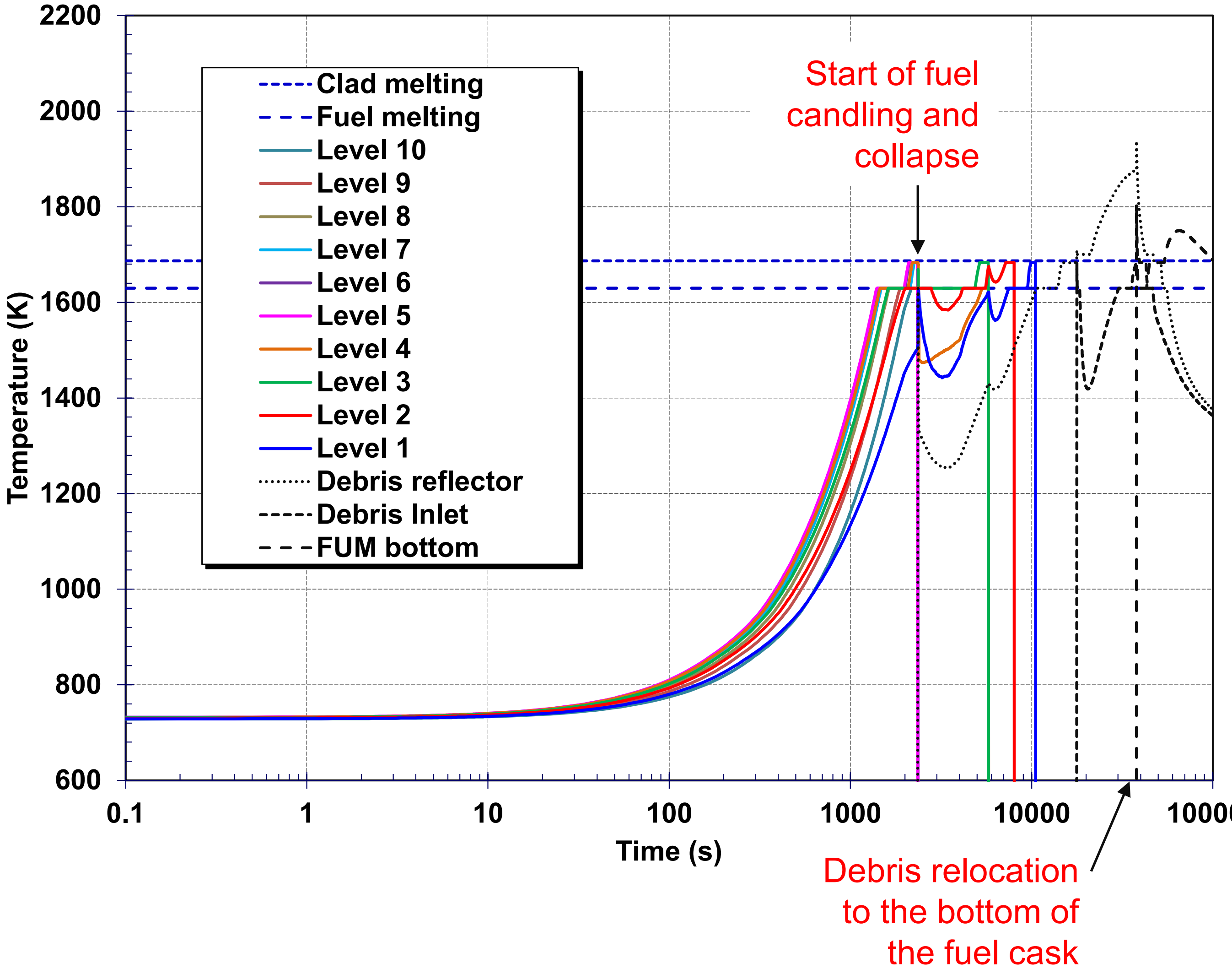
- During removal from the reactor, the fuel assemblies are blown dry with argon gas
  - No residual sodium was included in the accident scenario
- Fuel assemblies with normal in-vessel storage cool in the damaged FUM (i.e., very low decay heat)
- The accidental removal of a recently discharged assembly would lead to fuel failure after 40 min





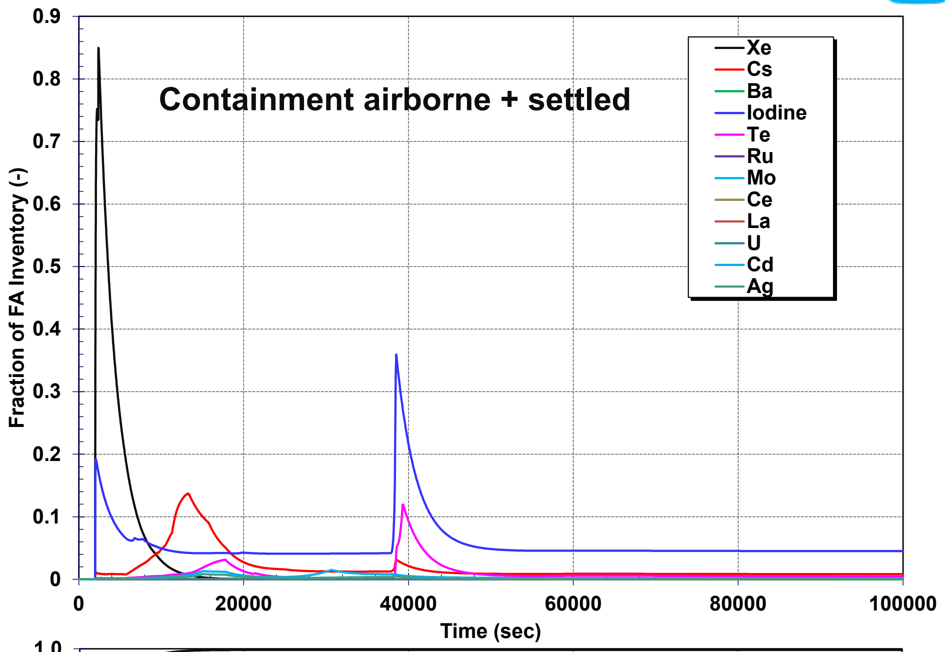
FUM accident scenario – recently discharged assembly results

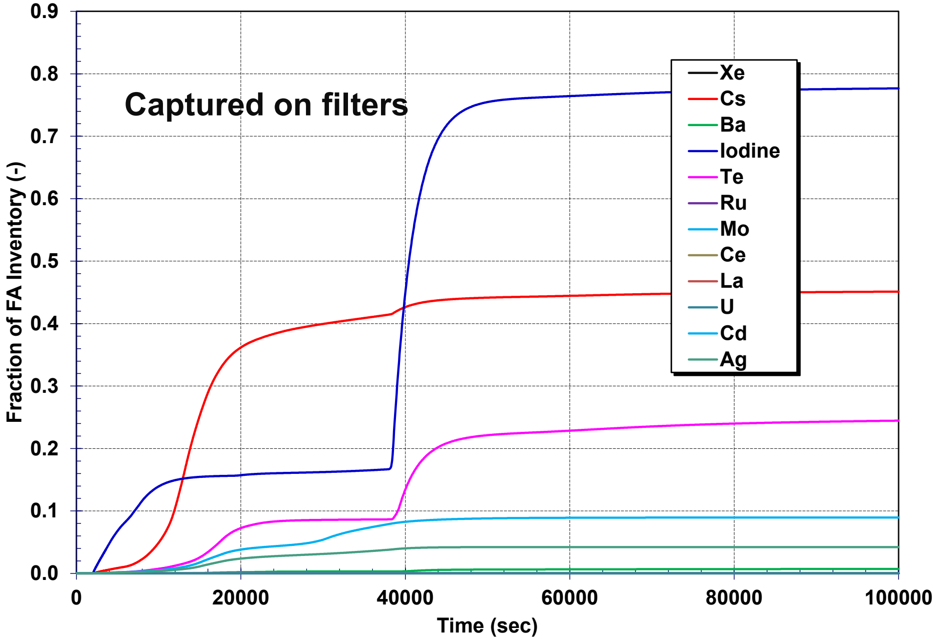
- If a recently discharged assembly is accidentally removed, it will rapidly heat to cladding candling and fuel rod failure conditions
- The assembly successively relocates downward to the bottom of the storage cask
- The high temperature fuel debris could fail the cask and spill out
  - Cask failure requires further design details
- Fission product release from the cask occurs through the assumed cracks after being dropped

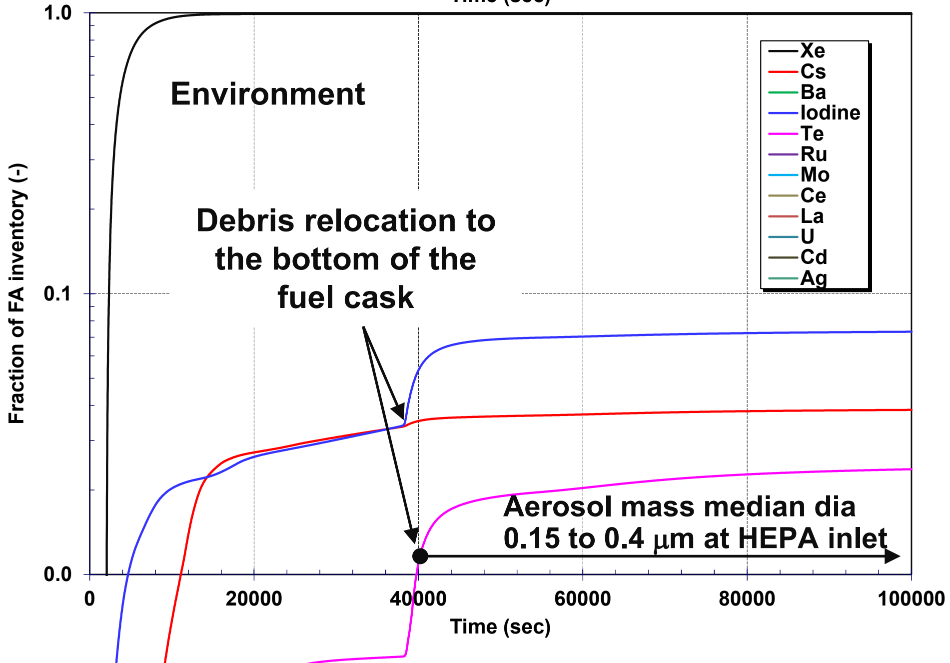


FUM accident scenario

- Noble gases were rapidly released from the FUM following the fuel degradation and vented to the environment
- Early release of more volatile cesium was captured on the filters
- CsI (and NaI) and Te primarily came out following the failure of the assembly inlet structure at 38,000 sec (10 hr)
- HEPA filter performance modeled to degrade below 0.3 μm diameter aerosols per typical HEPA specifications





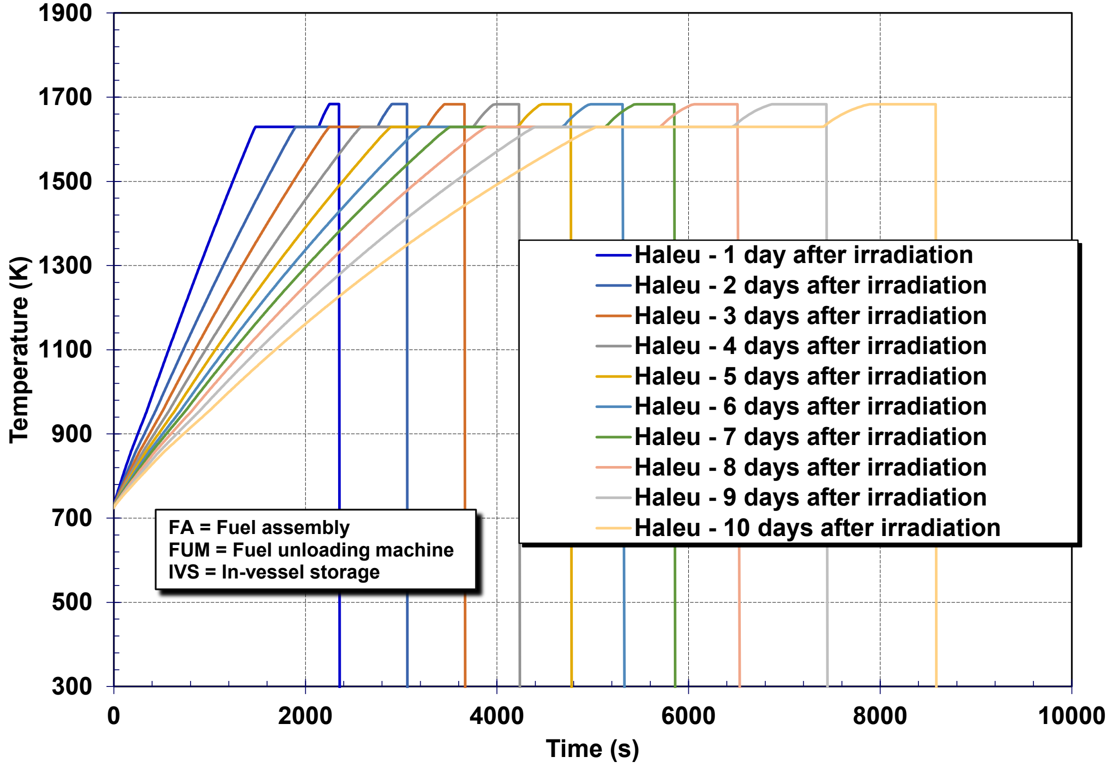


FUM accident scenario sensitivity calculations

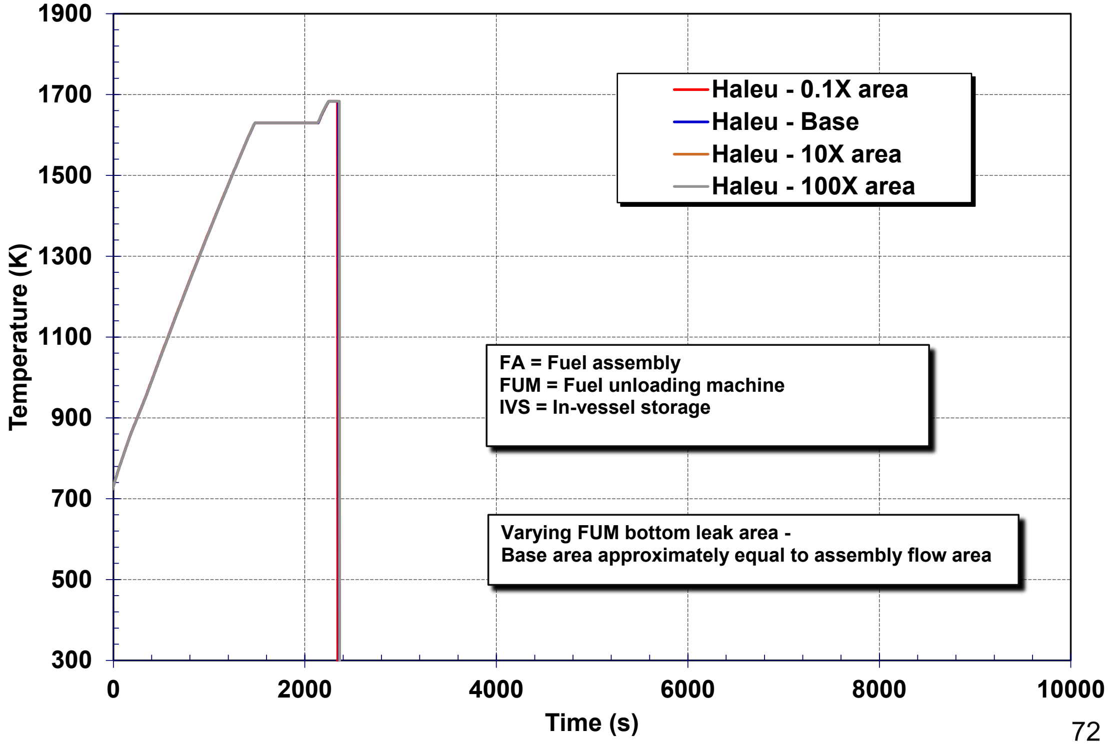
- The earliest timing of an assembly removal from the vessel was uncertain
- Fuel collapse started at 2360 sec (0.7 hr) with one day of cooling but increased to 8560 sec (2.4 hr) with 10 days of cooling

- Increasing the bottom leakage flow area had a negligible impact on the accident scenario progression
  - Convective cooling due to leakage had a negligible impact
  - The upper leakage path from the FUM was much larger than the assembly flow area
  - The base bottom leakage was equal to the assembly flow area.

Fuel temperature as a function of time after irradiation



Fuel temperature as a function of leakage area



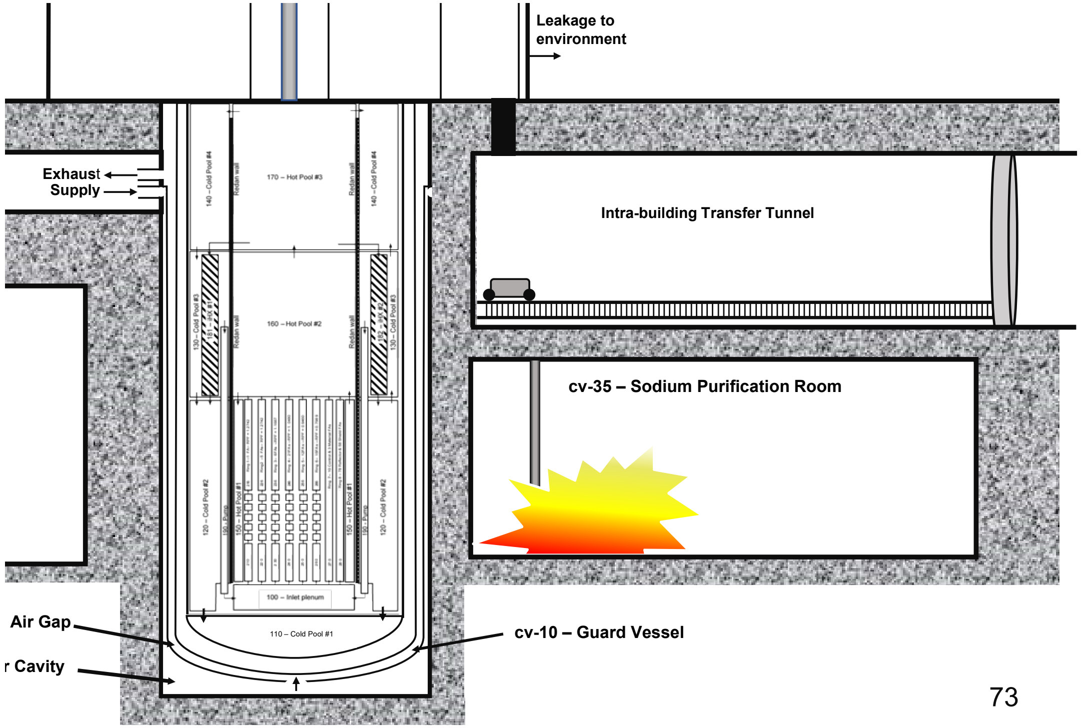
Sodium purification system pipe break scenario

The ABTR sodium purification system filters sodium from the reactor to remove hydrogen and oxygen impurities and monitors for crystallization and plugging indicators

- The inlet and exit piping penetrates through the reactor vessel enclosure (i.e., the vessel upper lid)
- The purification piping was specified as a 3” diameter pipe and assumed to break in the sodium purification room
- MELCOR predicted the sodium siphon flow to be 18 kg/s with vessel cover gas pressure of 0.3 bar and a full pipe break

The scenario includes failure of the cladding boundary on 217 fuel rods (i.e., 1 assembly)

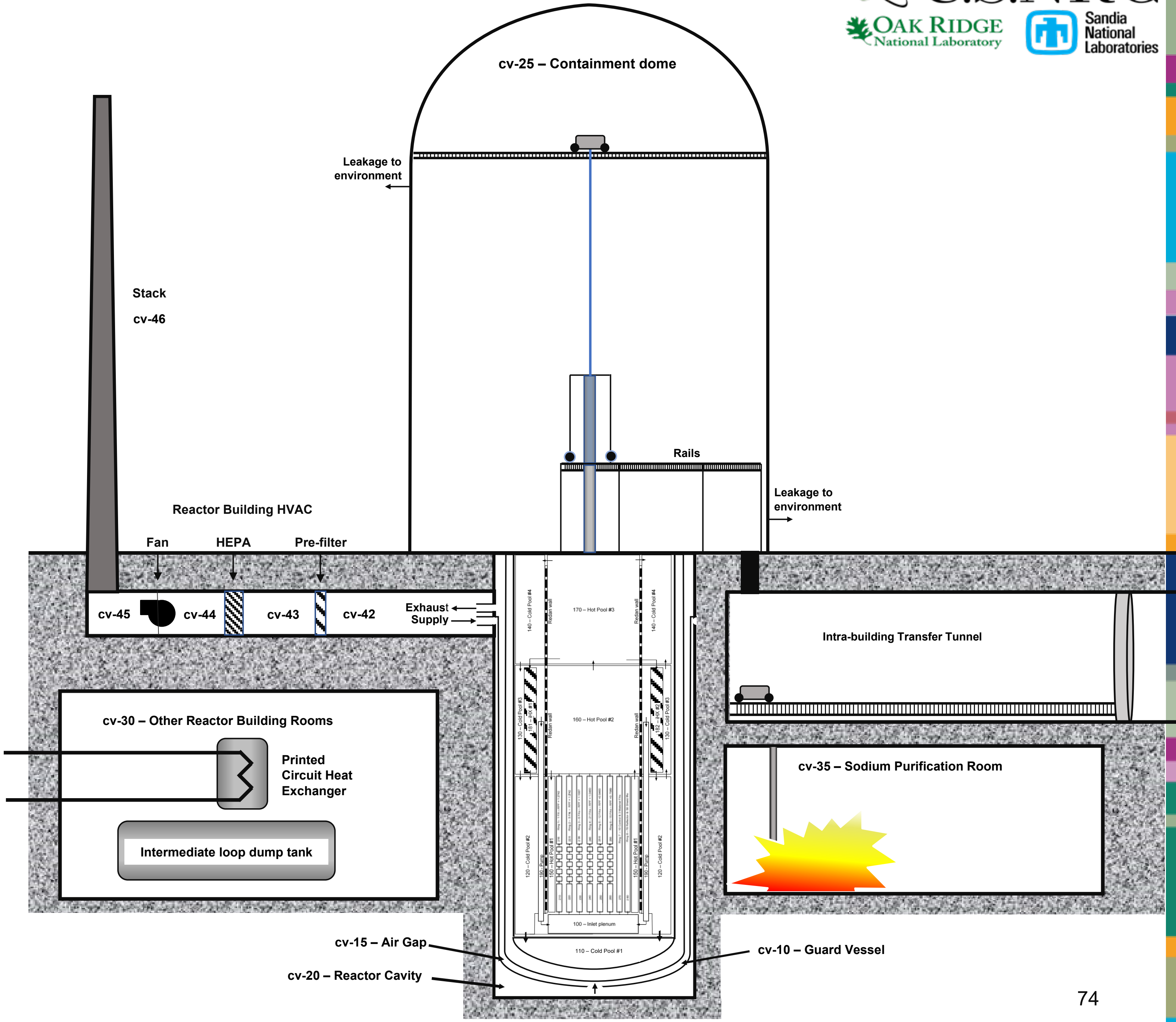
The reactor building HVAC system is operating with ~2X air-changes per hour to maintain a -2” H_2O gauge pressure in the sodium purification room



Sodium purification system pipe break scenario

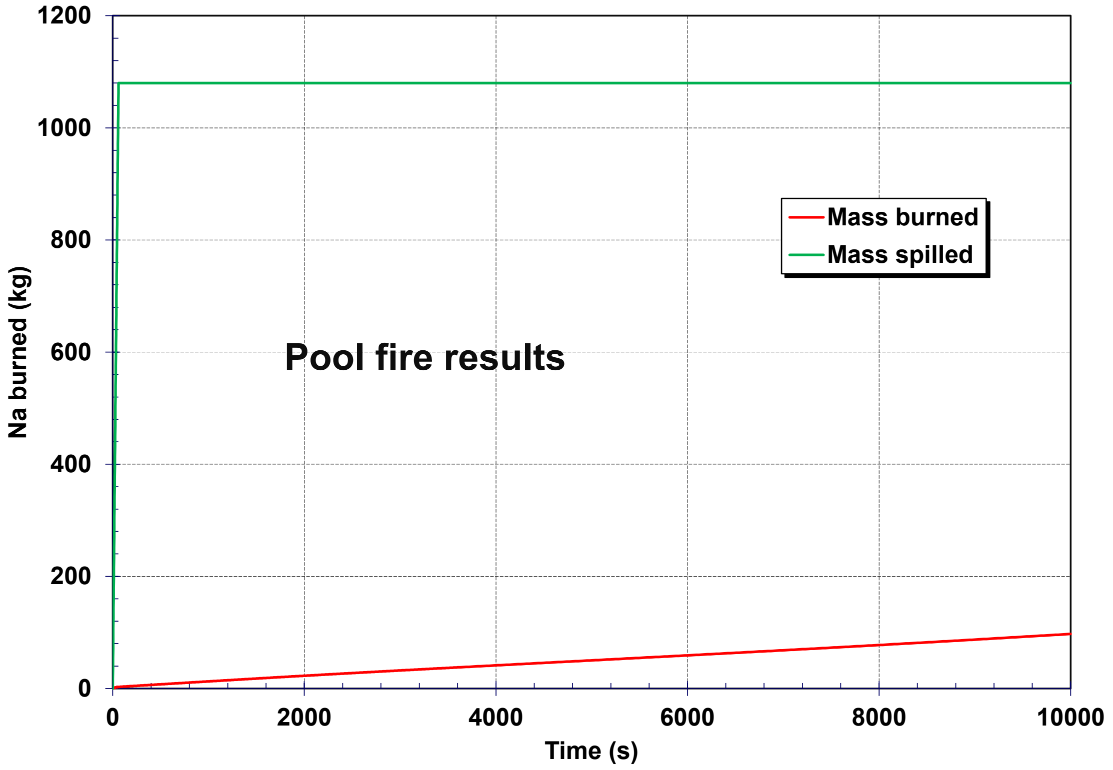
Accident scenario assumptions

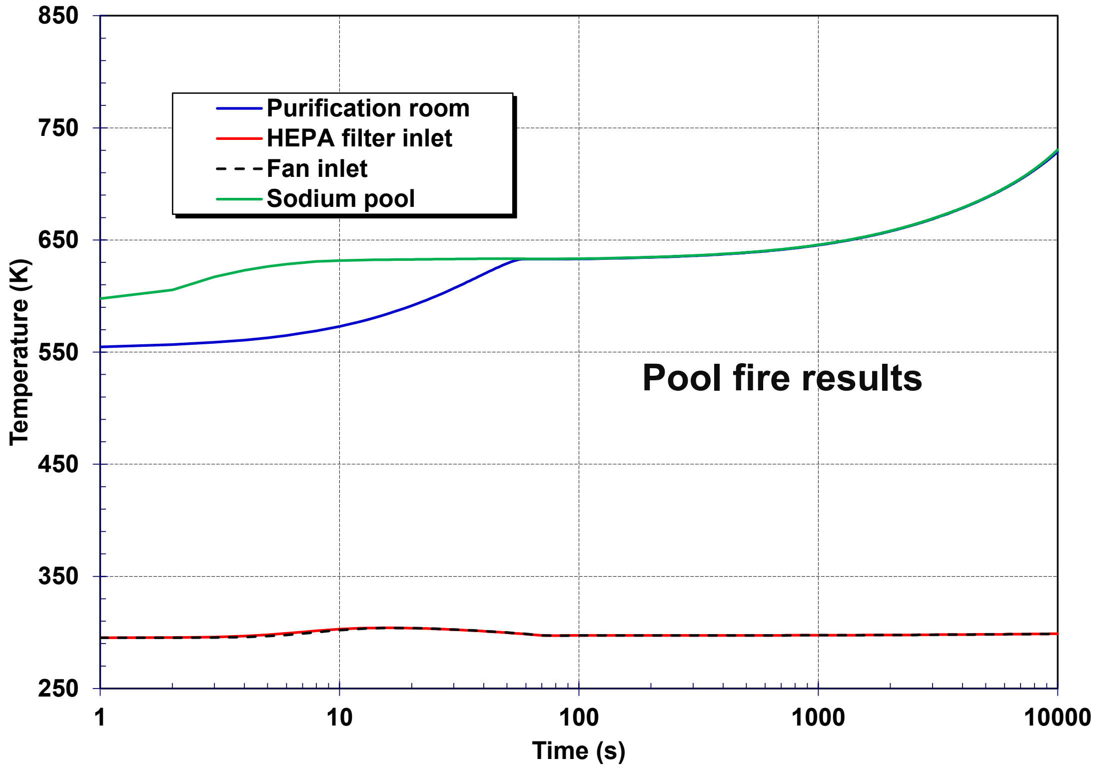
- Sodium piping is isolated at 60 sec (nominally)
- Pipe break is 1 m above the floor
- Siphon flow for full pipe break is 18 kg/s (varied)
- Spray droplet size varied
- Pool and spray+pool fire scenarios



Sodium purification system pipe break scenario

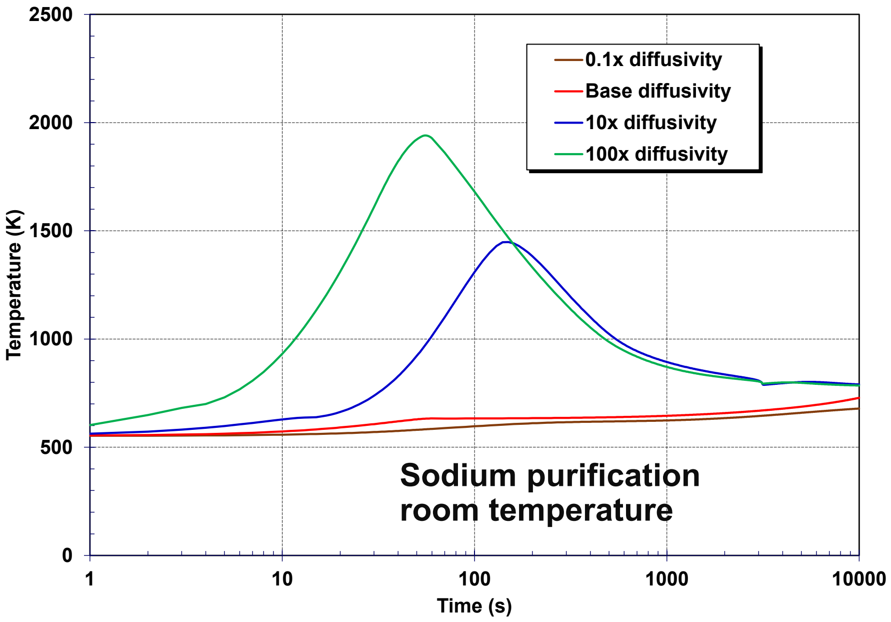
- 1080 kg of sodium spilled into the purification room before being isolated
  - Purification system isolated at 60 sec
- Pool fire scenario results below assume no spray oxidation and a maximum pool diameter of 3 m (i.e., room constraints)
- Oxide layer forms on the pool surface and limits oxygen diffusion into the pool (~10% burned in 2.8 hr)
- Pool will slowly burn for days without mitigation

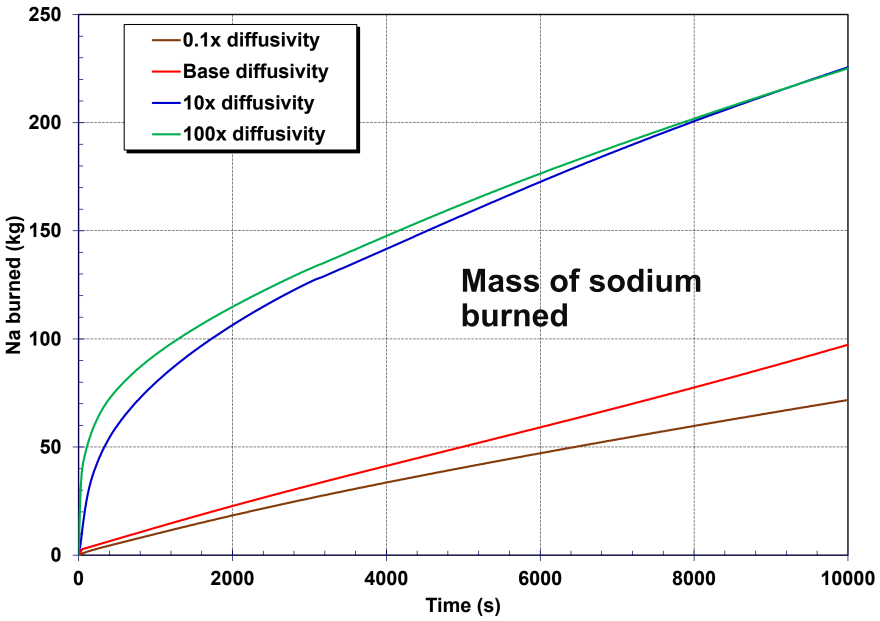




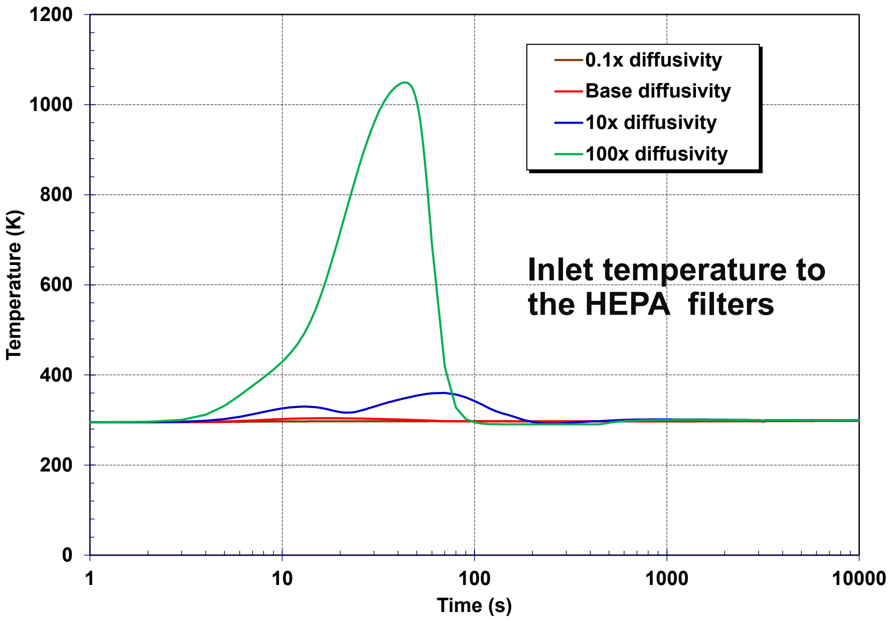
Sodium purification system pipe break scenario

- The sodium burn rate is controlled by the oxide layer on the pool surface
  - Oxide layer eventually builds up to limit the burn rate
- Oxygen diffusivity across the oxide layer on the pool surface has uncertainties, which initially affect the burn rate
  - e.g., pool geometry, pool temperature, room oxygen
  - Oxide layer eventually limits oxygen diffusivity
- The peak room temperature and the gas temperature to the HEPA filters is strongly impacted by the initial burn rate





Pool fire results



Sodium purification system pipe break scenario

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Oak Ridge National Laboratory
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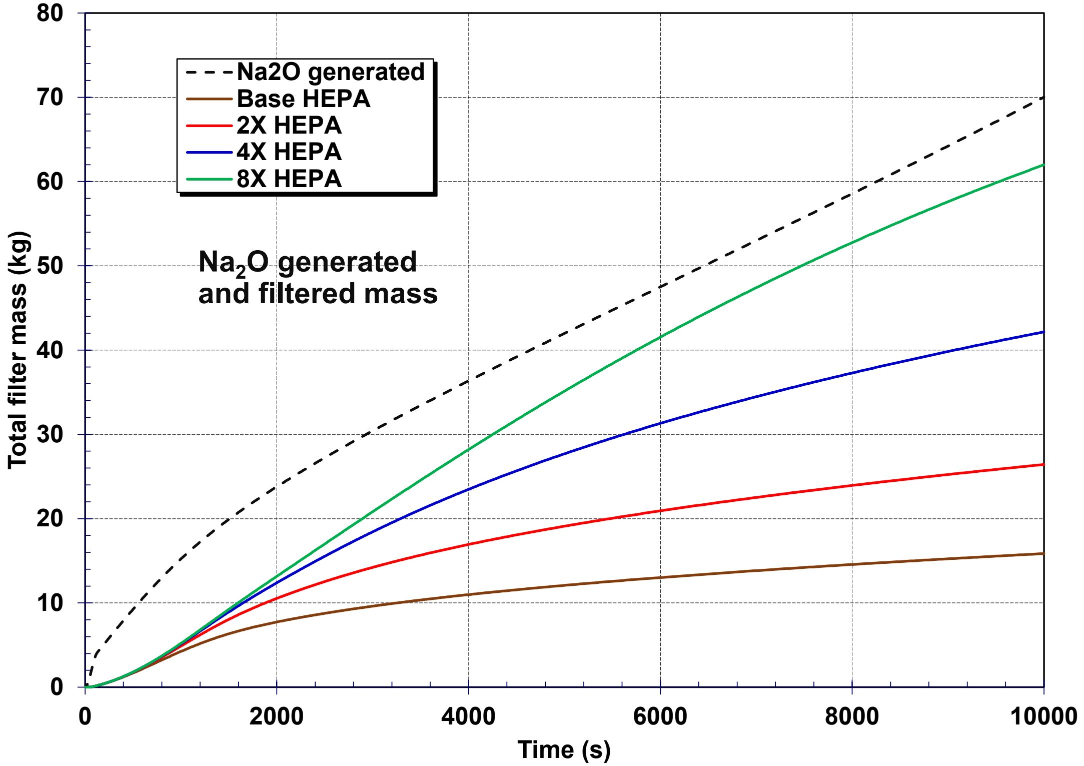
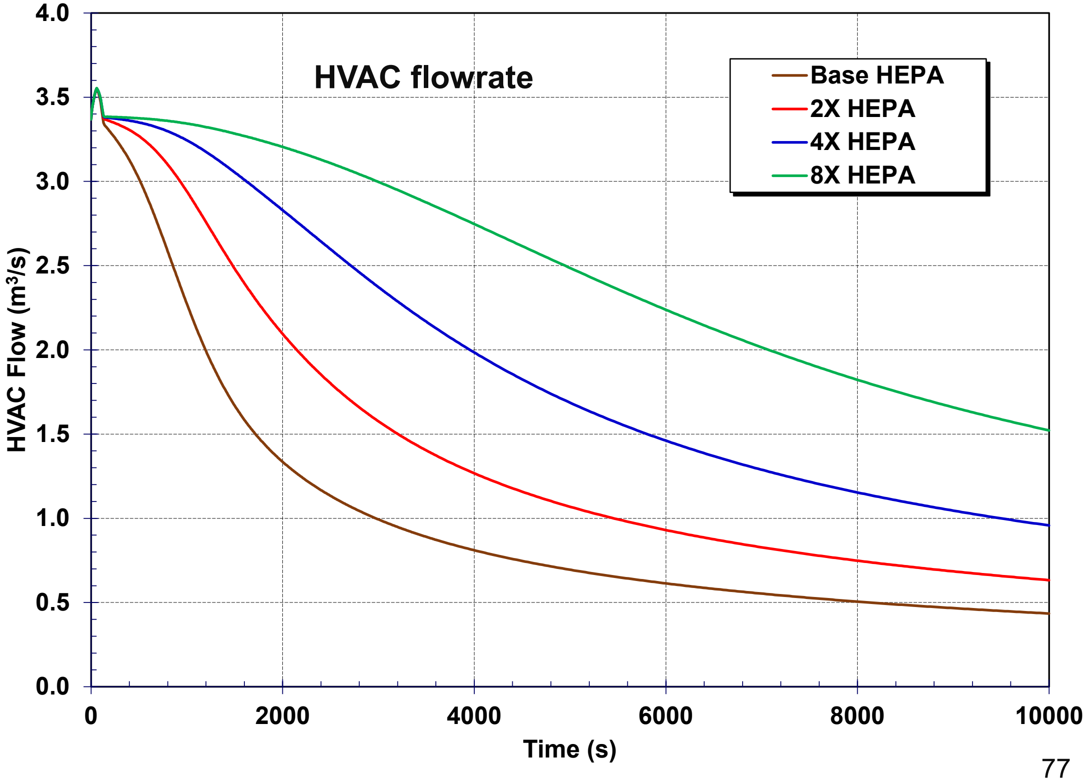
- Sodium fires generate lots of aerosols
  - $2\ Na + \frac{1}{2}\ O_2 \rightarrow Na_2O$ (dominant in these calculations)
- Sodium byproduct aerosols plug filters and reduce HVAC flow & effectiveness
  - Base case assumes 1 HEPA filter unit (i.e., not described in the ABTR reference report)
  - Sensitivity calculations assess the impact of 2, 4, and 8 HEPA filter units

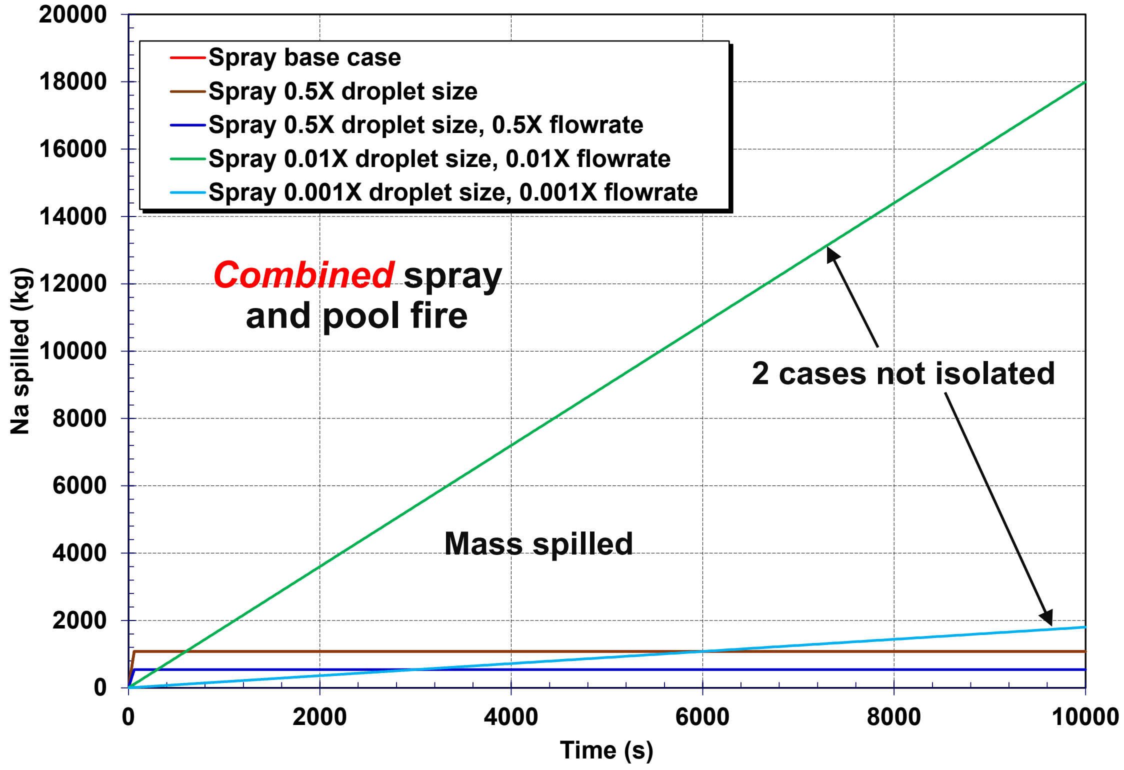


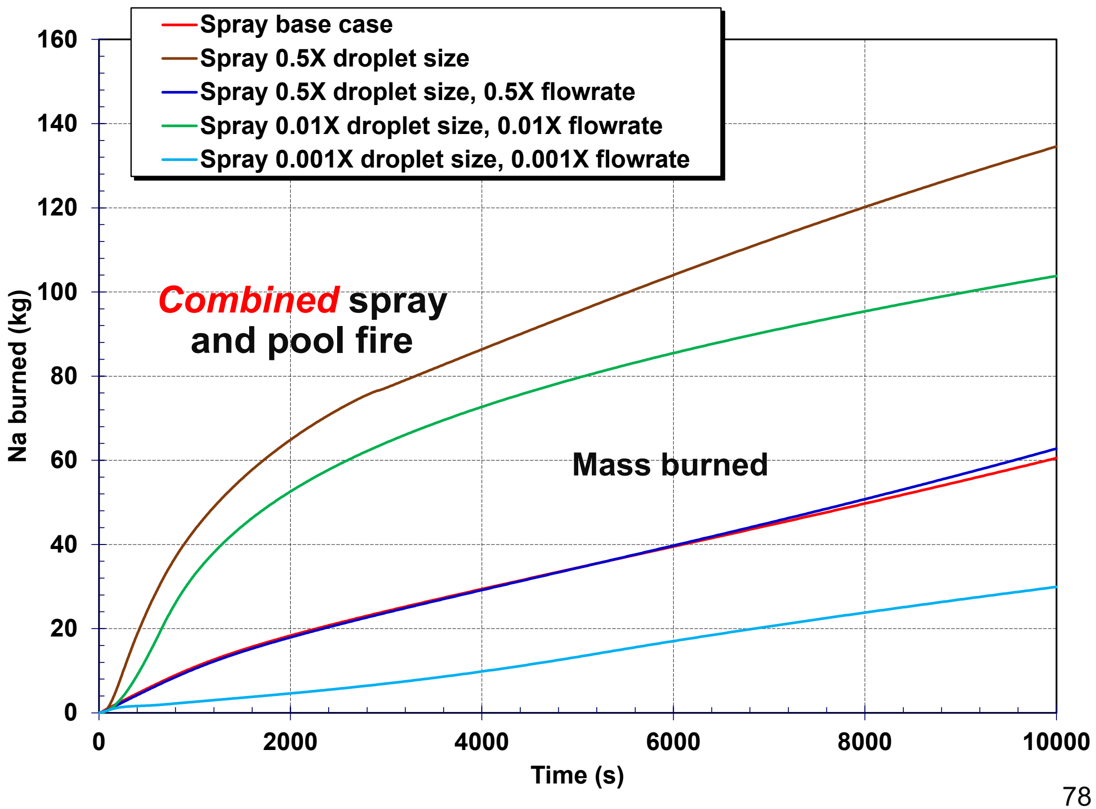
Pool fire results



Sodium purification system pipe break scenario

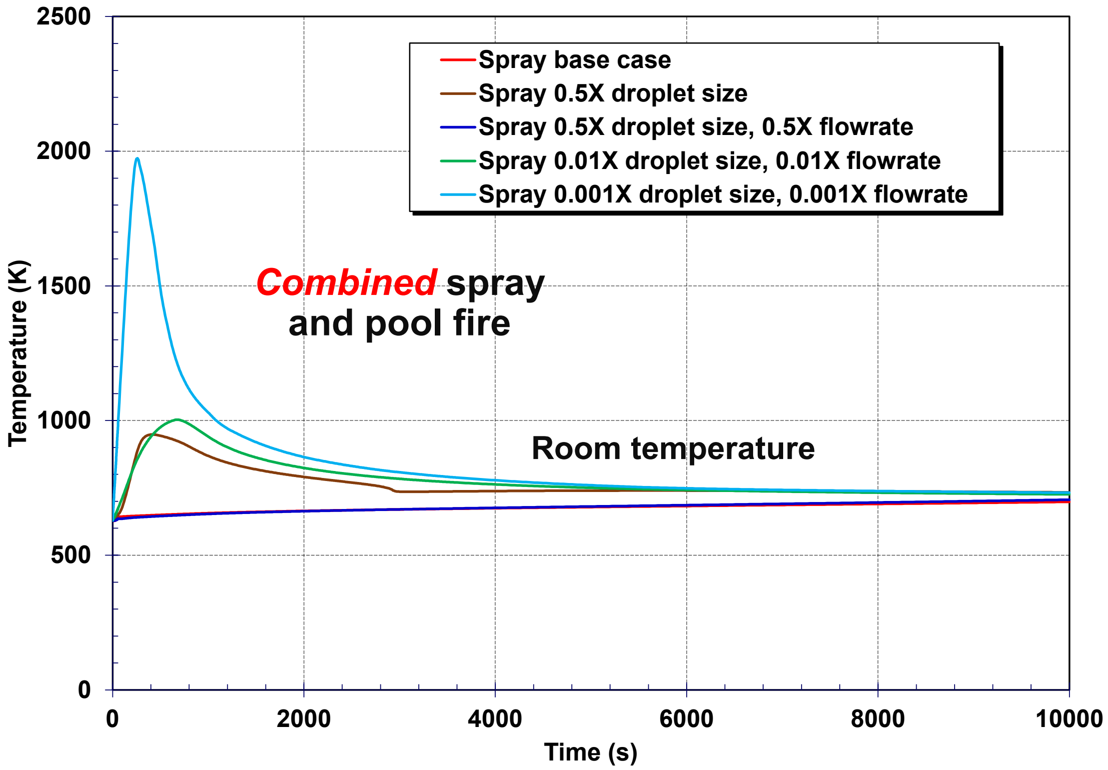
- Next examples include *combined* spray and pool fires
  - Includes spray interaction with the room oxygen with continuation in a pool fire
- Base case is 18 kg/s with a large droplet size (i.e., characteristic of low-pressure pour)
- Other cases explored smaller spray droplet sizes, smaller flowrates, and isolated or not isolated
  - Mass burned is a function of droplet size, leak rate, and leak duration

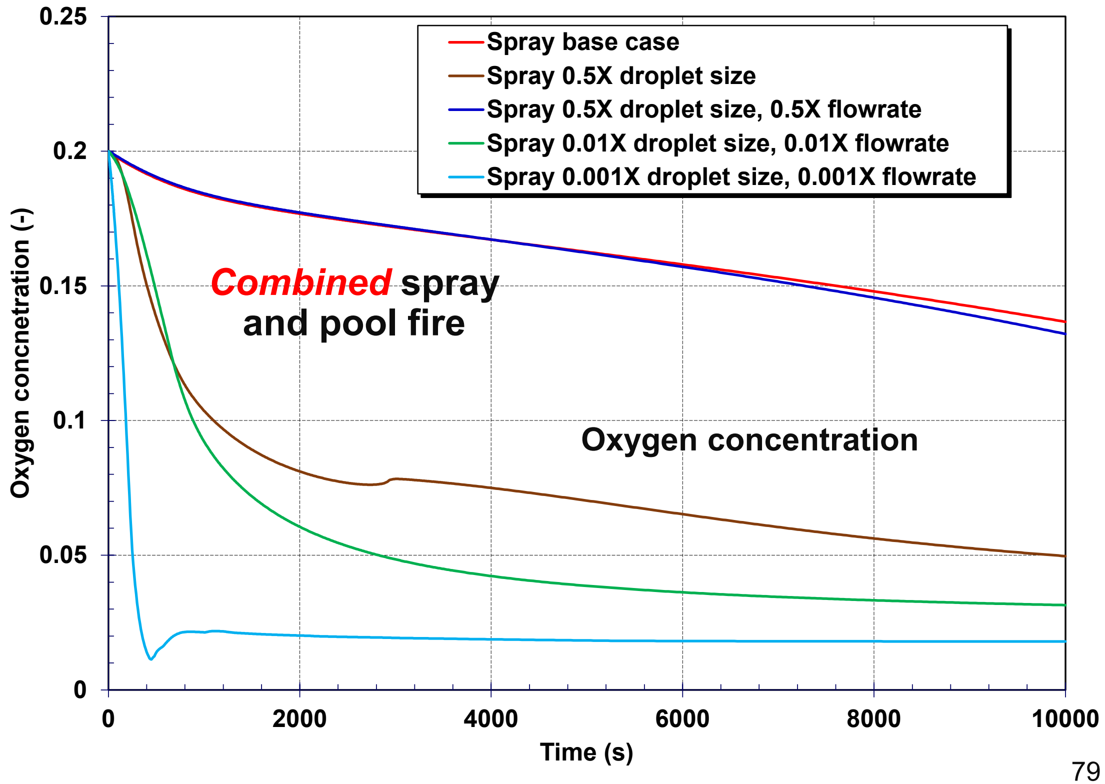




Sodium purification system pipe break scenario

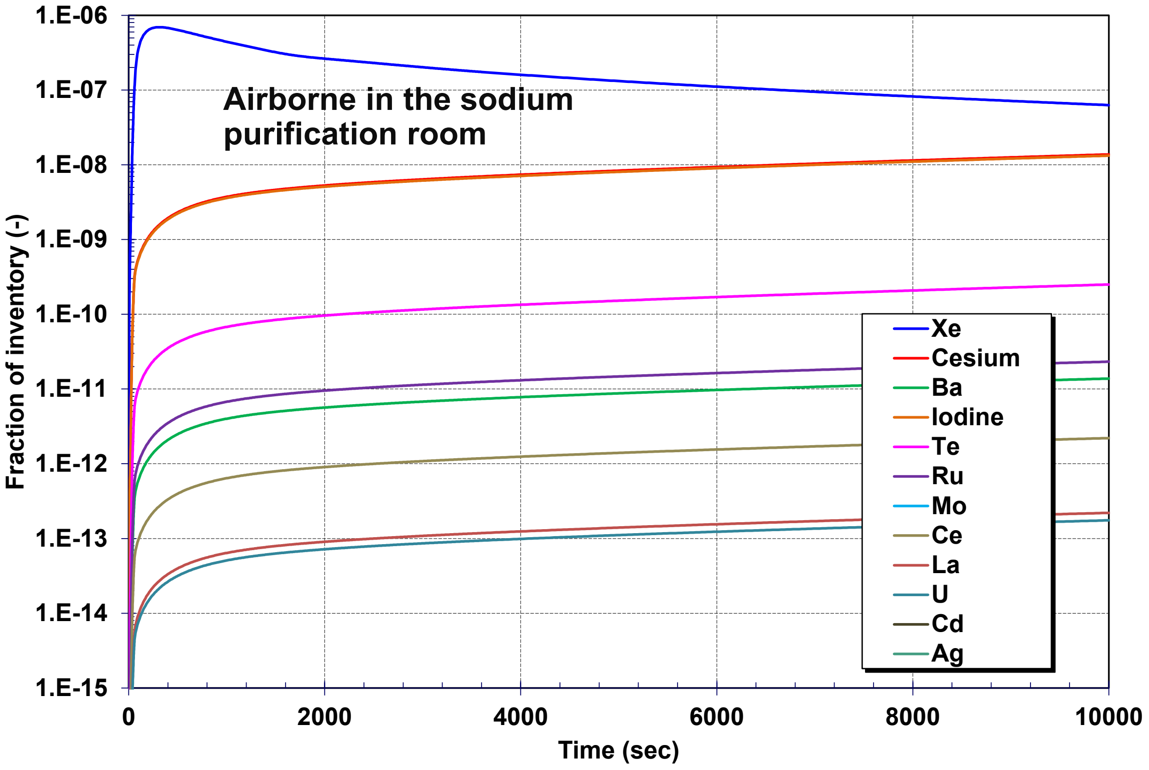
- Spray fire room temperatures can be much higher due to the spray burn efficiency versus a pool fire (i.e., function of droplet size, fall height, spray velocity)
- Sodium fires can be oxygen limited (HVAC remains operational)
  - Contrast the 0.001X spray droplet results at 0.001X mass flow rate with base case response

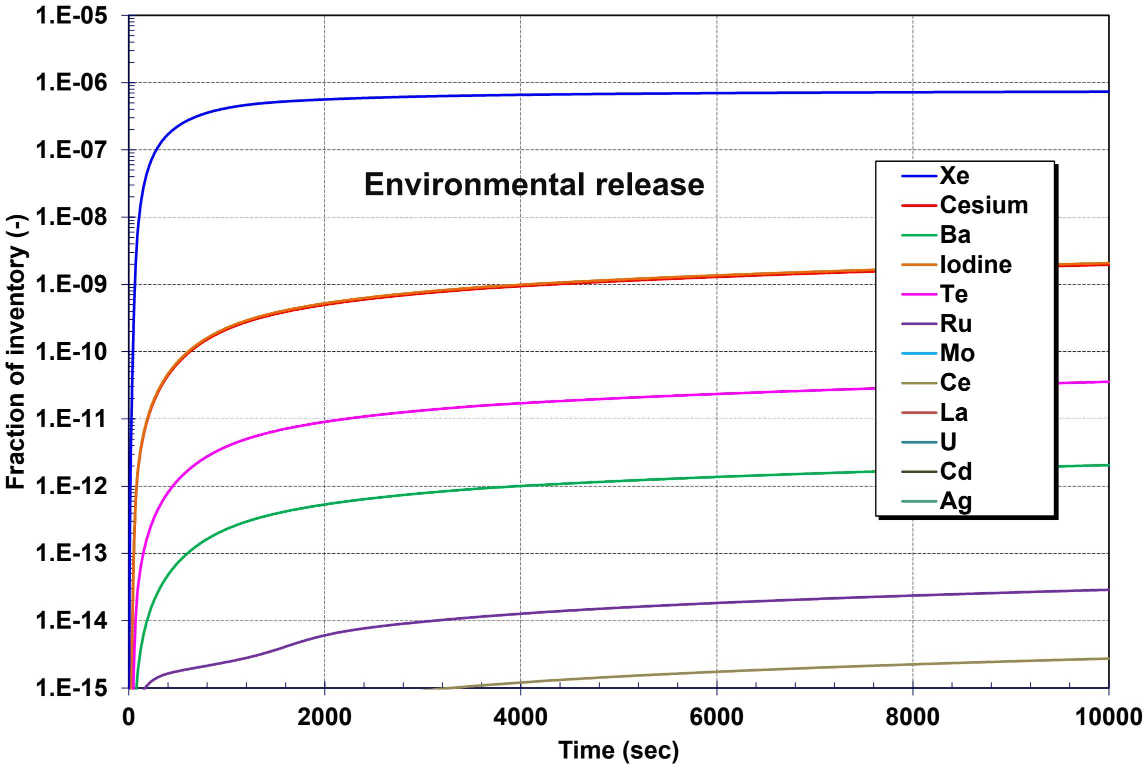




Sodium purification system pipe break scenario

- Release magnitude is limited by (a) the small amount of radionuclide inventory in the spill and (b) the slow burning rate (i.e., release rate is proportional to burn rate)
- The airborne concentration steadily decreases due to HVAC flow (initially 2 room changes per hour)
- HEPA filter captures most radionuclides and limits environmental release

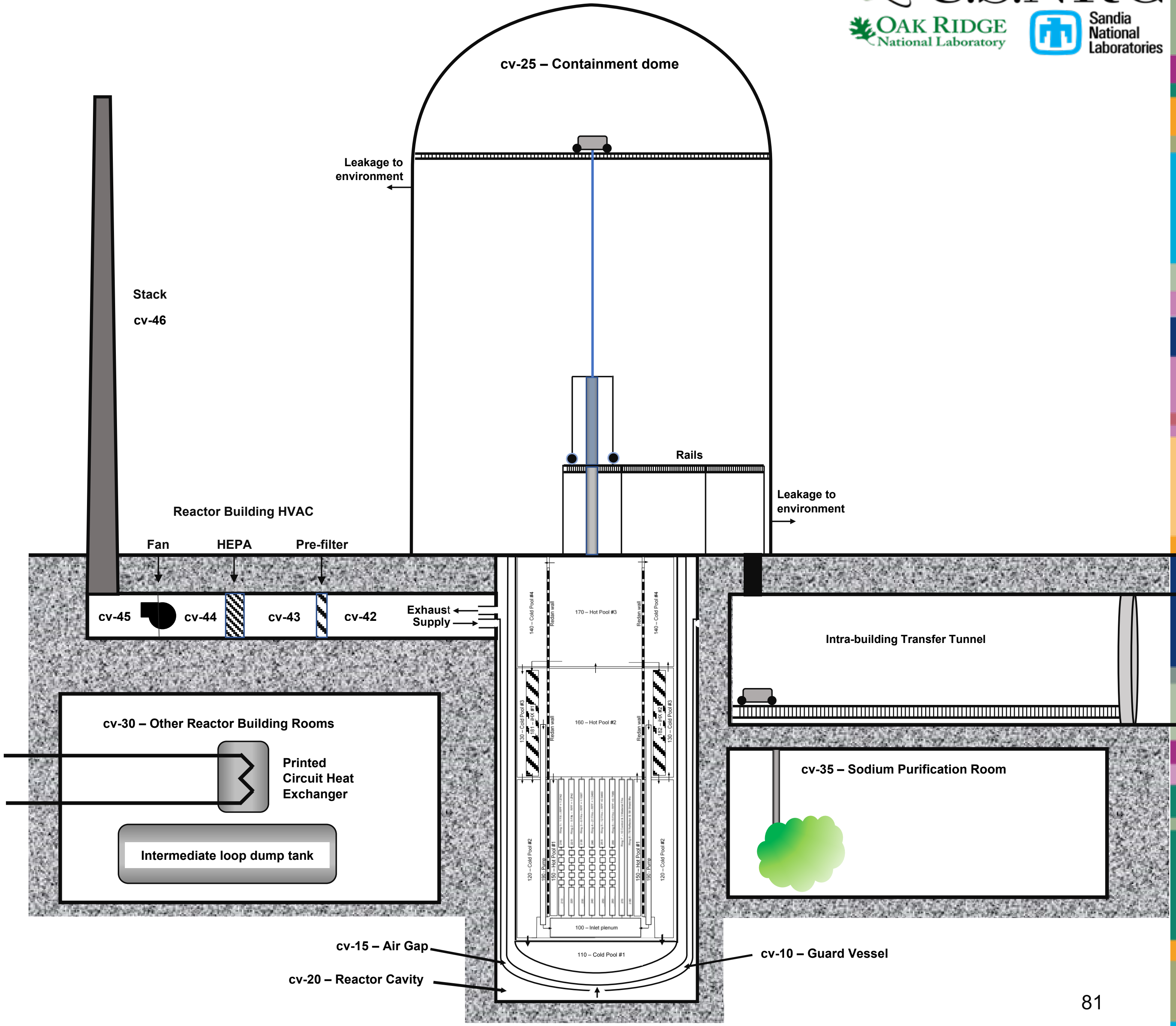




Cover-gas pipe break scenario

Accident scenario assumptions

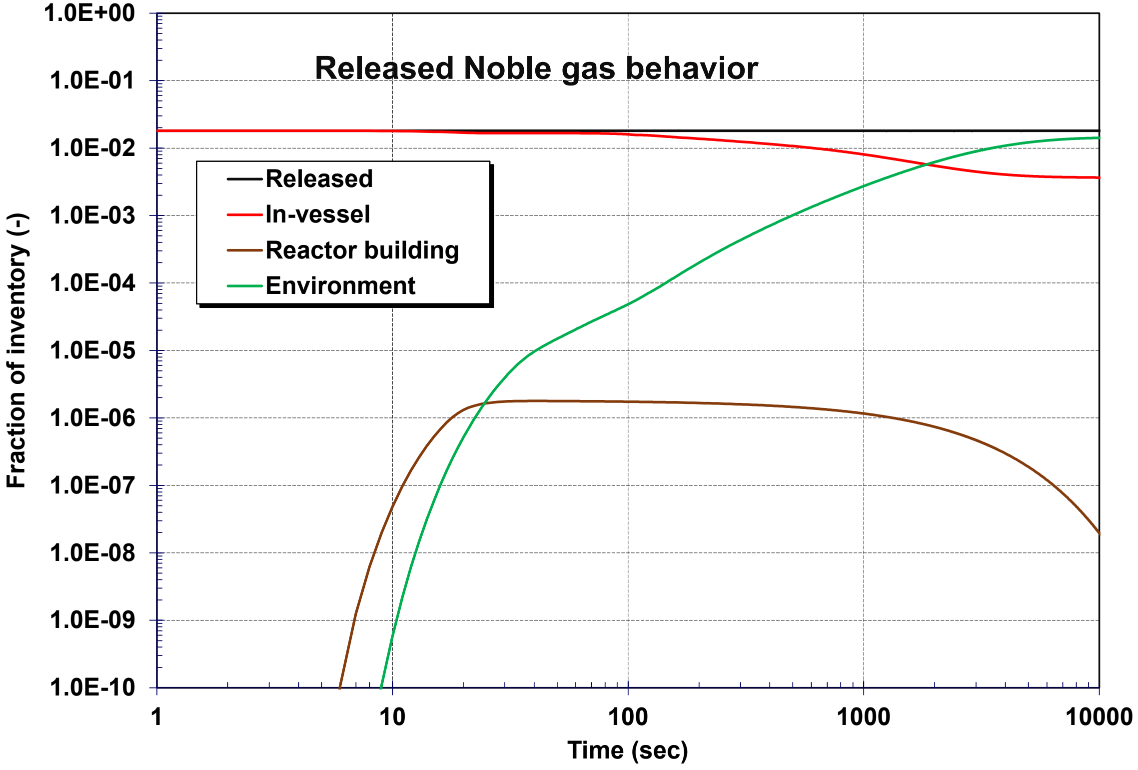
- Cover-gas piping is not isolated
- Discharge flow is steady and maintained by large pressure control supply tanks
- HVAC is running with 2X air changes per hour
- The scenario includes failure of the cladding boundary on 217 fuel rods (i.e., 1 assembly)

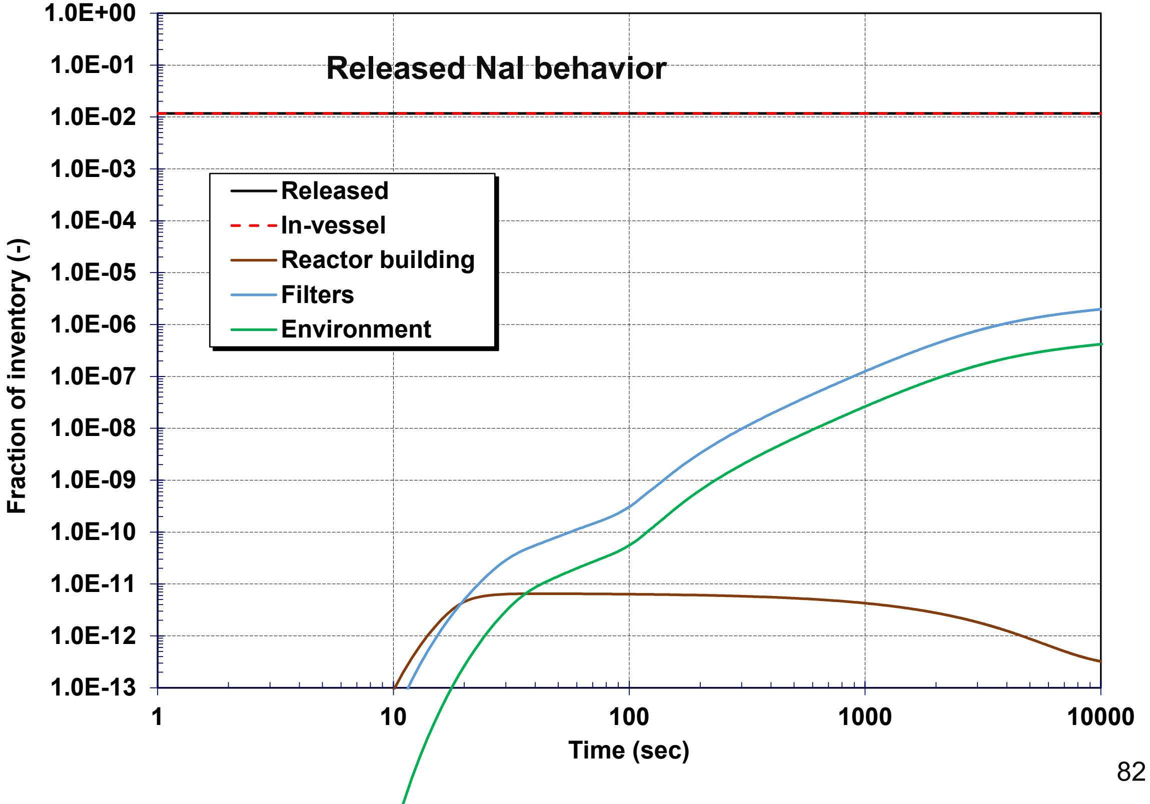


Cover-gas pipe break scenario

- The noble gases released from the failed fuel claddings circulate with the sodium but eventually rise to the surface of the sodium pool
- Once in the cover gas, they leak through the cover gas pipe break.
- The HVAC circulates the released gases out the plant stack

- The released iodine combines with sodium to form sodium iodine (NaI).
- Most of the NaI remains in the pool due to its low vapor pressure in this scenario (~0.01 Pa)
- The released NaI condenses into small aerosols that are not completely filtered by the HEPA



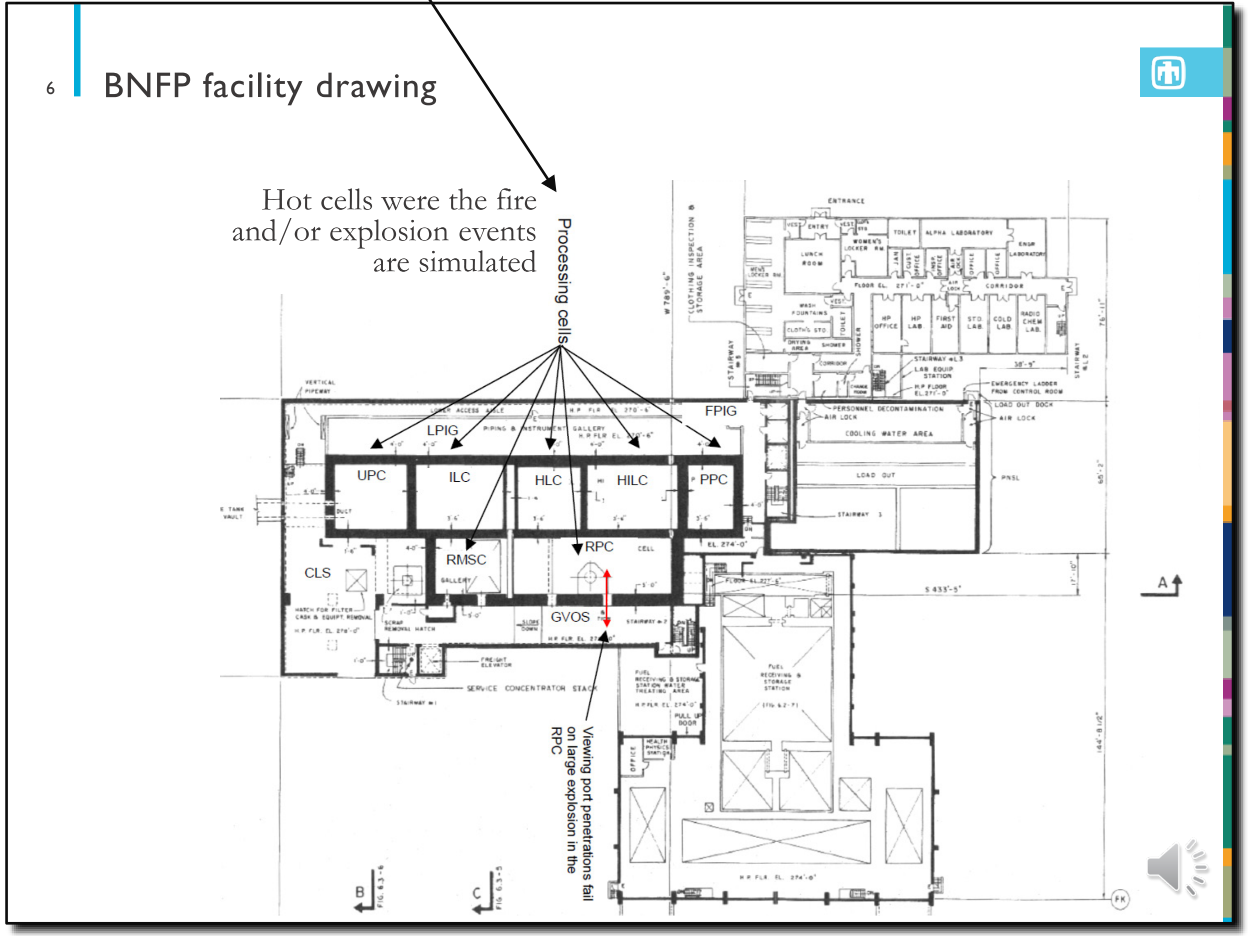


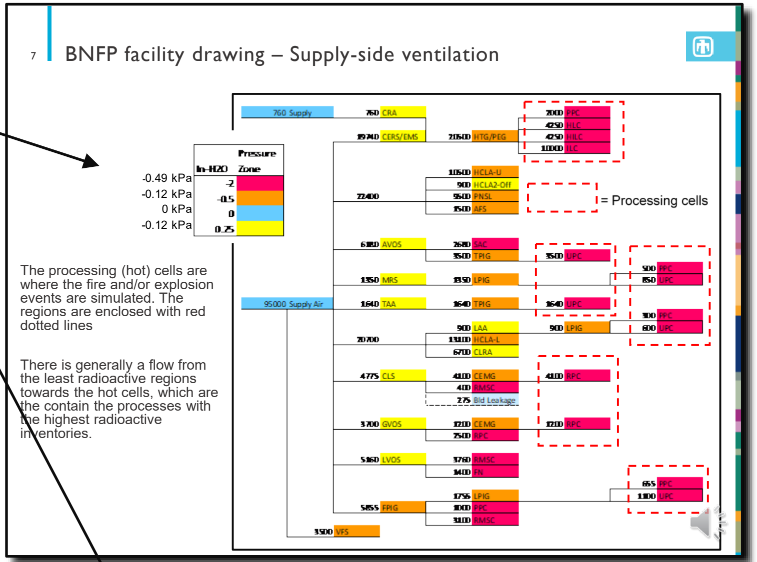
Examples for fuel fabrication and reprocessing safety analysis

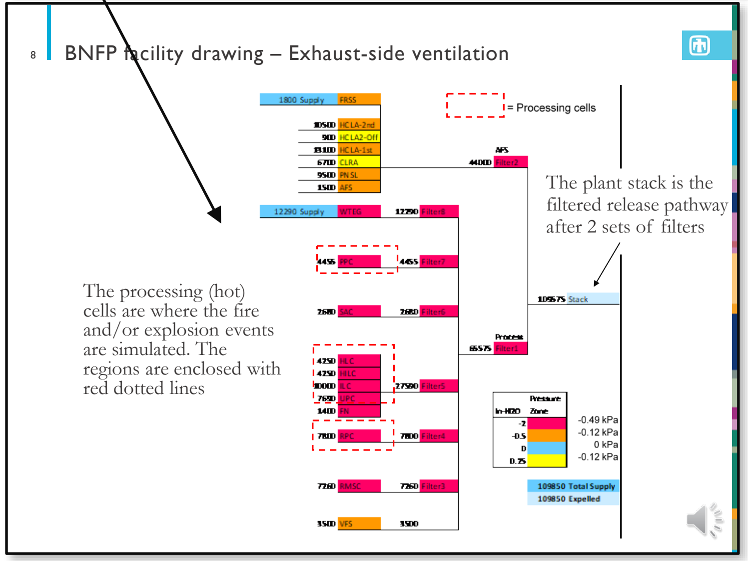
Reprocessing and fuel fabrication accident analysis

Hot cells for hazardous material processing

Safety-grade ventilation and filtration system

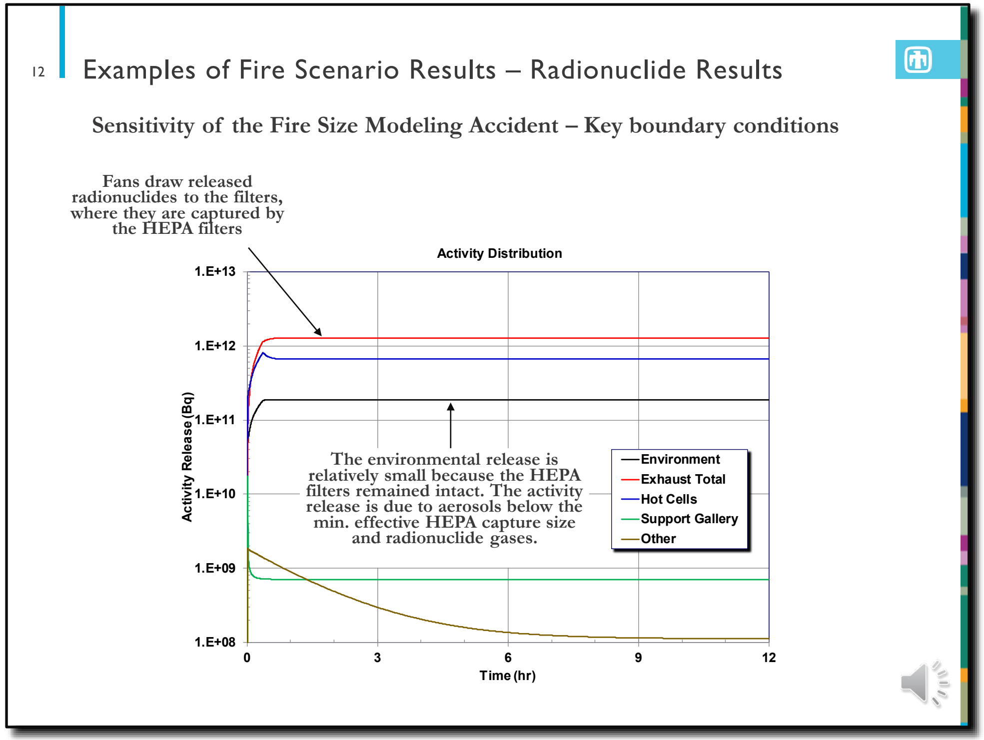






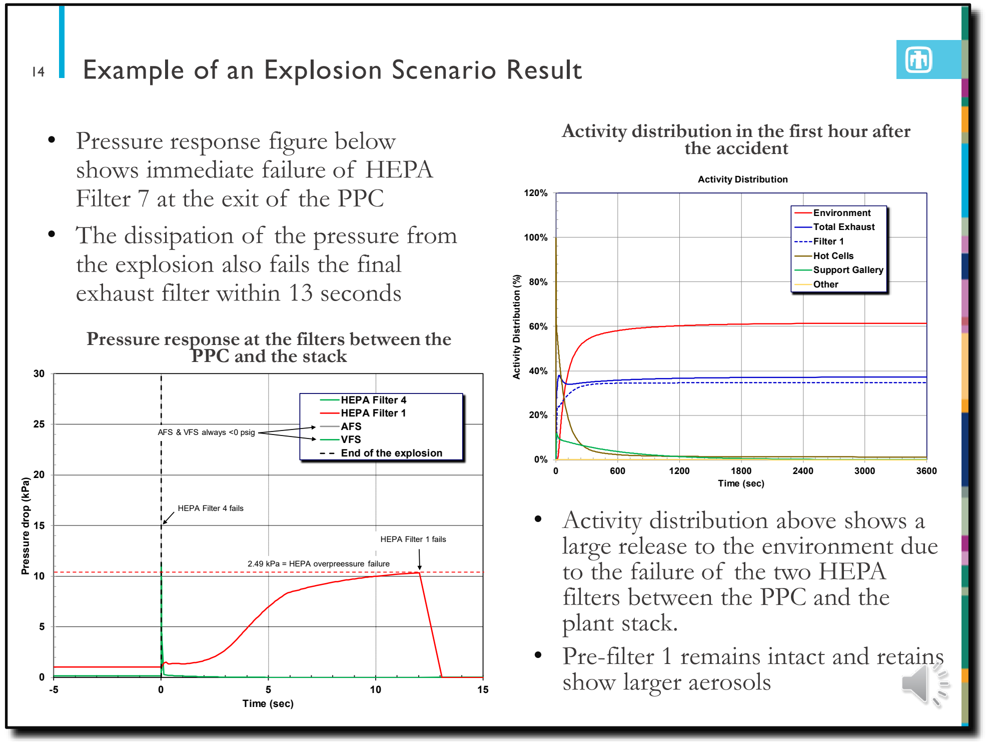
Ref. [K. C. Wagner and David L.Y. Louie, "MELCOR Demonstration Analysis Of Accident Scenarios At A Spent Nuclear Reprocessing Plant," 28th International Conference on Nuclear Engineering, August 2- 6, 2020, Anaheim, CA, USA, ICONE28-POWER2020-16584]

Reprocessing and fuel fabrication accident analysis



Example of a fire scenario

Example of an explosion scenario



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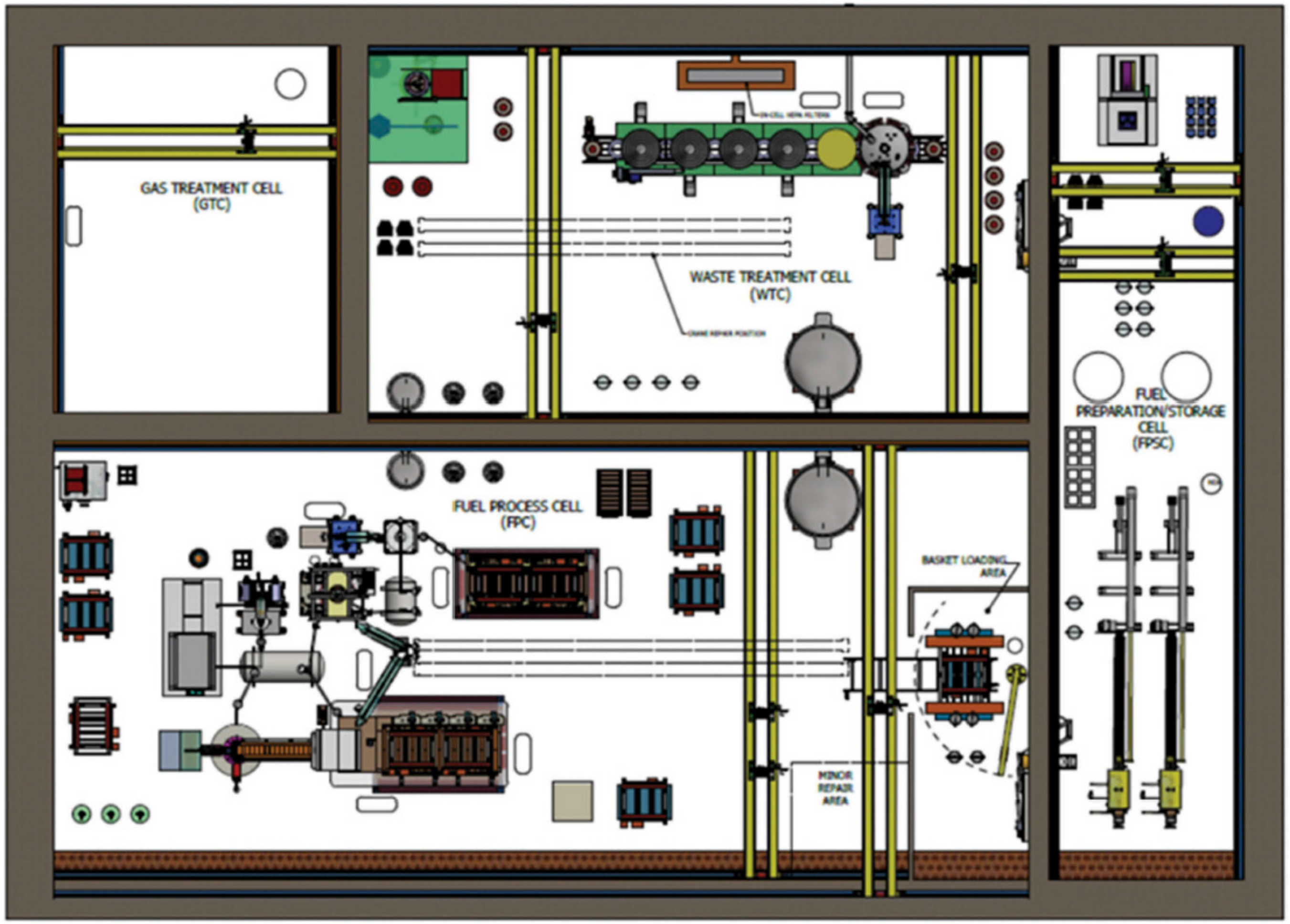
Reprocessing and fuel fabrication accident analysis

Argonne National Laboratory and Merrick & Company, Engineering Services recently published a concept for a pyro-processing plant

- Insufficient information for a demonstration calculation
- Similar to the Barnwell facility, work done in hot cells
- Cited *limiting* accident with oxidation of 1000-2000 kg of uranium metals
- Other accidents due to loss of heat removal for TRU vault
- Fuel fabrication could include spill accidents during casting and alloying steps



[Yoon Il Chang, et al. (2018): Conceptual Design of a Pilot-Scale Pyroprocessing Facility, Nuclear Technology https://doi.org/10.1080/00295450.2018.1513243]

MELCOR Summary

MELCOR SFR Summary

- MELCOR capabilities were demonstrated
  - New phenomenological modeling added to MELCOR for SFRs
  - Application of radionuclide transport models
- Capabilities for a range of SFR fuel cycle accident scenarios
- Key physics considered
  - SFR assembly thermal hydraulics
  - Sodium fires
  - Fission product release
- Future work
  - Fission product release modeling from spills and sodium fires
  - Radionuclide chemistry

Workshop Summary

Closing Remarks

- *Demonstration of NRC's Code Readiness for Simulating non-LWRs*
  - *HTGR Nuclear Fuel Cycle (Completed February 2023)*
  - *SFR Nuclear Fuel Cycle (Today)*

- *Next Steps*
  - *Public Reports*
    - *Coming in 2023, "Non-LWR Fuel Cycle Scenarios for SCALE and MELCOR Modeling Capability Demonstration"*
  - *MSR Nuclear Fuel Cycle Workshop (2024)*

Backup

IAEA –TECDOC-2006 Notes

<u>CDA no mixing release insights</u>

Halogens

- NaI(l) is predominant chemical species for real mixture
- Bromine forms CsBr with 50% release at 950 K (no mixture) but drops to <10^{-4} with mixing

Alkali metals

- Cs binds to CsI, CsRb. CsBr, CsNa with 90% release
- Complete Rb release

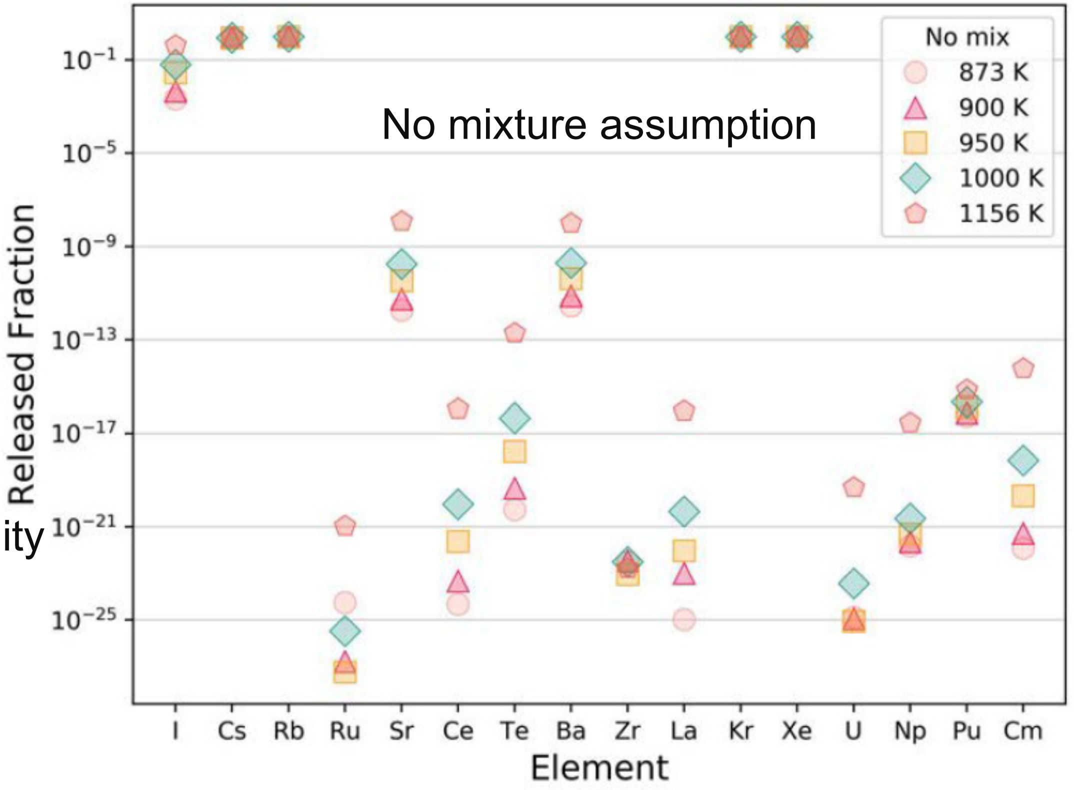
Tellurium

- BaTe which does not release

Others

- Noble metals are solid & do not release
- Lanthanides form oxides and dependent on oxygen availability
- Eu is volatile (13% release) if it does not form Eu_2O_3
- Ce, Pu, and Np are stable

- No mixture → compound vapor pressure
- Ideal mixture → Raoult's Law
- Real mixture → Excess for deviation from Raoult's Law



IAEA –TECDOC-2006 Notes

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873 K