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?

- What system(s) are we analyzing?
- What code(s) are we using?
- What are the key phenomena being considered?
- Are there any gaps in modeling capabilities of the selected codes? How do we close these gaps?
- What data do we have & what data do we need?

IAP Strategy 2
Volume 5
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**

- Enrichment (UF₆ enrichment)
- UF₆ Transportation
- Fuel Fabrication
- Fresh Fuel Transportation
- 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 High 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
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 – UF₆ enrichment
T1 – Transportation of UF₆ 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

LWR Operation

Reprocessing & Fabrication

Utilization

R1 – Reprocessing
F1 – Fuel fabrication
F2 – Fuel assembly fabrication
T2 – Transportation of assemblies/salt to plant
U1 – Fresh fuel staging/preparation/loading

○ Considered in this work
○ Not considered in this work

Disposal

S1

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

Scenario for this stage studied in this workshop
E1: Enrichment

• Enrichment of UF₆ up to 19.75 wt.% ²³⁵U [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.% ²³⁵U 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 UF₆ HALEU
      - Phase 2 (1 year): production of 900 kg UF₆ HALEU
      - Phase 3 (3 year): production of 900 kg UF₆ HALEU/year

Major hazards:
• UF₆ 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:

<table>
<thead>
<tr>
<th>Package design</th>
<th>Enrichment limit</th>
<th>Permissible UF$_6$ mass</th>
</tr>
</thead>
<tbody>
<tr>
<td>DN30-10</td>
<td>10 wt.% $^{235}$U</td>
<td>1460 kg</td>
</tr>
<tr>
<td>DN30-20</td>
<td>20 wt.% $^{235}$U</td>
<td>1271 kg</td>
</tr>
</tbody>
</table>

- 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

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

Major hazards:
- Criticality from misfeeding or mishandling of fuel
- Release of radiological materials

Refs.:
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 UF₄ 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

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

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

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

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

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)

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
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.

**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)

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**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.

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**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.

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**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.

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*Ref.: Pyroprocessing Technologies Brochure, Argonne National Laboratory*
REFERENCE SODIUM FAST REACTOR DESIGN

Advanced Burner Test Reactor (ABTR)

<table>
<thead>
<tr>
<th>Reactor Power</th>
<th>250 MWt, 95 MWe</th>
</tr>
</thead>
<tbody>
<tr>
<td>Coolant Temperature</td>
<td>355°C/510°C</td>
</tr>
<tr>
<td>Fuel</td>
<td>Metallic</td>
</tr>
<tr>
<td>Cladding and Duct</td>
<td>HT-9</td>
</tr>
<tr>
<td>Cycle Length</td>
<td>4 months</td>
</tr>
</tbody>
</table>

Refs.:
## APPLIED SCALE 6.3.1 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)

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

### 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

Ref:
# SPENT NUCLEAR FUEL

## ABTR TRU Fuels

Source terms for all scenarios (ABTR TRU Inner)

<table>
<thead>
<tr>
<th>Source</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>U/TRU-10Zr</td>
<td>16.5 wt.% (inner) &amp; 20.7 wt.% TRU (outer)</td>
</tr>
<tr>
<td>Specific power</td>
<td>65.6 GW/tHM (inner) &amp; 51.4 GW/tHM (outer)</td>
</tr>
<tr>
<td>Discharged BU</td>
<td>94.5 GWD/tHM (inner) &amp; 92.6 GWD/tHM (outer)</td>
</tr>
</tbody>
</table>

## ABTR HALEU Fuel

Source terms for all scenarios

<table>
<thead>
<tr>
<th>Source</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>U-10Zr</td>
<td>16.5 wt.% U-235</td>
</tr>
<tr>
<td>Specific power</td>
<td>46.2 GW/tHM</td>
</tr>
<tr>
<td>Discharged BU</td>
<td>149.74 GWD/tHM</td>
</tr>
</tbody>
</table>

## PWR Fuel

Source terms for scenarios 2 and 3

<table>
<thead>
<tr>
<th>Source</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>UO₂</td>
<td>4.95 wt.% U-235</td>
</tr>
<tr>
<td>Specific power</td>
<td>33.7 GW/tHM</td>
</tr>
<tr>
<td>Discharged BU</td>
<td>50.00 GWD/tHM</td>
</tr>
</tbody>
</table>

Refs.:

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)
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.
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)

<table>
<thead>
<tr>
<th>Fuel</th>
<th>At 10 days of cooling time</th>
<th>At 5 years of cooling time</th>
</tr>
</thead>
<tbody>
<tr>
<td>U/TRU Inner</td>
<td>140La (21%) 106Rh (12%) 144Pr (9%) 95Nb (8%) 95Zr (8%)</td>
<td>137mBa (22%) 106Rh (14%) 90Y (12%) 238Pu (9%) 134Cs (7%)</td>
</tr>
<tr>
<td>U/TRU Outer</td>
<td>140La (21%) 106Rh (12%) 144Pr (9%) 95Nb (8%) 95Zr (8%)</td>
<td>137mBa (22%) 106Rh (12%) 90Y (12%) 238Pu (10%) 134Cs (6%)</td>
</tr>
<tr>
<td>HALEU</td>
<td>140La (21%) 144Pr (11%) 95Nb (9%) 95Zr (9%) 106Rh (7%)</td>
<td>90Y (29%) 137mBa (29%) 134Cs (11%) 137Cs (7%) 238Pu (6%)</td>
</tr>
<tr>
<td>PWR*</td>
<td>140La (21%) 144Pr (10%) 95Nb (8%) 106Rh (8%) 95Zr (8%)</td>
<td>90Y (25%) 137mBa (25%) 238Pu (11%) 244Cm (11%) 137Cs (7%)</td>
</tr>
</tbody>
</table>
SPENT NUCLEAR FUEL - ACTIVITY

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

<table>
<thead>
<tr>
<th>Fuel</th>
<th>At 10 days of cooling time</th>
<th>At 5 years of cooling time</th>
</tr>
</thead>
<tbody>
<tr>
<td>U/TRU Inner</td>
<td></td>
<td></td>
</tr>
<tr>
<td>103Ru (8%)</td>
<td>103mRh (8%)</td>
<td>137Cs (19%)</td>
</tr>
<tr>
<td>103mRh (8%)</td>
<td></td>
<td>137mBa (18%)</td>
</tr>
<tr>
<td>95Nb (7%)</td>
<td></td>
<td>241Pu (15%)</td>
</tr>
<tr>
<td>95Zr (7%)</td>
<td></td>
<td>147Pm (12%)</td>
</tr>
<tr>
<td>141Ce (6%)</td>
<td></td>
<td>90Y (7%)</td>
</tr>
</tbody>
</table>

| U/TRU Outer |                          |                            |
| 103Ru (8%)  | 103mRh (8%)              | 137Cs (19%)                |
| 103mRh (8%) |                        | 241Pu (19%)                |
| 95Nb (7%)   |                          | 137mBa (18%)               |
| 95Zr (6%)   |                          | 147Pm (11%)                |
| 141Ce (6%)  |                          | 90Y (7%)                   |

| HALEU |                          |                            |
| 95Nb (8%) |                          | 137Cs (22%)                |
| 95Zr (7%)  |                          | 137mBa (21%)               |
| 103Ru (6%)  |                          | 90Y (16%)                  |
| 103mRh (6%)  |                          | 95Sr (16%)                 |
| 141Ce (6%)  |                          | 147Pm (10%)                |

| PWR* |                          |                            |
| 140La (32%) |                          | 137mBa (76%)                |
| 95Nb (14%)  |                          | 134Cs (16%)                 |
| 95Zr (13%)  |                          | 154Eu (7%)                  |
| 103Ru (9%)  |                          | 125Sb (0.5%)                |
| 134Cs (6%)  |                          | 108Rh (0.2%)                |

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

- 1.2-cm thick steel liner
- Reinforced concrete (~1 m) assuming rebar-to-concrete mass ratio of 0.106
- Fuel assembly

3D view of the CB with front quarter segment removed

Refs.:
Neutron sources from spontaneous fission

- Fuel light element impurities might contribute additional neutron sources

<table>
<thead>
<tr>
<th>Cooling time</th>
<th>ABTR HALEU</th>
<th>ABTR U/TRU (inner)</th>
</tr>
</thead>
<tbody>
<tr>
<td>10 days</td>
<td>Cm-242 (74.2%)</td>
<td>Cm-242 (44.3%)</td>
</tr>
<tr>
<td></td>
<td>Pu-240 (17.3%)</td>
<td>Cm-244 (54.0%)</td>
</tr>
<tr>
<td>7 cycles</td>
<td>Pu-240 (71.3%)</td>
<td>Cm-244 (94.1%)</td>
</tr>
<tr>
<td></td>
<td>Pu-238 (16.2%)</td>
<td>Cm-242 (0.27%)</td>
</tr>
<tr>
<td></td>
<td>Cm-244 (11.5%)</td>
<td></td>
</tr>
</tbody>
</table>

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

- 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

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

- 7 cycles of cooling

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Half-life</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sr-90/Y-90</td>
<td>28.78 yr/2.67 d</td>
</tr>
<tr>
<td>Ru-106/Rh-106</td>
<td>1.02 yr/2.18 h</td>
</tr>
<tr>
<td>Ag-110m</td>
<td>249.8 d</td>
</tr>
<tr>
<td>Sb-125</td>
<td>2.758 yr</td>
</tr>
<tr>
<td>Cs-134</td>
<td>2.065 yr</td>
</tr>
<tr>
<td>Cs-137/Ba-137m</td>
<td>30.07 yr/2.552 m</td>
</tr>
<tr>
<td>Ce-144/Pr-144</td>
<td>284.6 d/17.28 m</td>
</tr>
<tr>
<td>Eu-152</td>
<td>13.54 yr</td>
</tr>
<tr>
<td>Eu-154</td>
<td>8.593 yr</td>
</tr>
</tbody>
</table>
DOSE RATE MAP INSIDE CB

10 days cooling time

ABTR HALEU

Dose rate (rem/h)

- 7.0×10² rem/h (7.0 Sv/h)
- 4.6×10⁶ rem/h (4.6×10⁴ Sv/h)

ABTR U/TRU

Dose rate (rem/h)

- 9.0×10² rem/h (9.0 Sv/h)
- 6.0×10⁶ rem/h (6.0×10⁴ Sv/h)

Scale

- Logarithmic
DOSE RATE MAP INSIDE CB

7 cycles of cooling time

ABTR HALEU

- Dose rate: $9.2 \times 10^4$ rem/h ($9.2 \times 10^2$ Sv/h)
- Activity: 13.5 rem/h (0.135 Sv/h)

ABTR U/TRU

- Dose rate: $1.9 \times 10^5$ rem/h ($1.9 \times 10^3$ Sv/h)
- Activity: 30 rem/h (0.3 SV/h)
10 days cooling time

DOSE RATE MAPS OUTSIDE CB

ABTR HALEU

Dose rate (mrem/h)

- 530 mrem/h (5.3 mSv/h)
- 0.4 mrem/h (4 μSv/h)

Fuel assembly

ABTR U/TRU

Dose rate (mrem/h)

- 720 mrem/h (7.2 mSv/h)
- 0.5 mrem/h (5 μSv/h)

Fuel assembly

Scale

- Logarithmic
DOSE RATE MAPS OUTSIDE CB

7 cycles of cooling time

ABTR HALEU

0.34 mrem/h (3.4 μSv/h)

0.2 μrem/h (2.0E-03 μSv/h)

Fuel assembly

ABTR U/TRU

6.6 mrem/h (66 μSv/h)

5 μrem/h (5E-02 μSv/h)

Fuel assembly

Dose rate (mrem/h)

- 3.73E+0 - 1.00E+1
- 1.39E+0 - 3.73E+0
- 5.18E-1 - 1.39E+0
- 1.93E-1 - 5.18E-1
- 7.20E-2 - 1.93E-1
- 2.68E-2 - 7.20E-2
- 1.00E-2 - 2.68E-2
- 3.73E-3 - 1.00E-2
- 1.39E-3 - 3.73E-3
- 6.18E-4 - 1.39E-3
- 1.93E-4 - 5.18E-4
- 7.20E-5 - 1.93E-4
- 2.68E-5 - 7.20E-5
- 1.00E-5 - 2.68E-5
- 3.73E-6 - 1.00E-5
- 1.39E-6 - 3.73E-6
- 5.18E-7 - 1.39E-6
- 1.93E-7 - 5.18E-7
- 7.20E-8 - 1.93E-7
- 2.68E-8 - 7.20E-8
- 1.00E-8 - 2.68E-8

Scale

- Logarithmic
For comparison, the irradiation dose of PWR spent fuel (50 GWd/tHM) after 10 days of cooling is about $1.7 \times 10^6$ rem/h ($1.7 \times 10^4$ 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 EBR II spent nuclear fuel treatment.
- The chopped PWR spent fuel will 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

ELECTROREFINING

ELECTROREFINING


CRITICALITY ANALYSIS OF ELECTROREFINER

CSAS Model of Electrorefiner (40”x40”)

Single Cathode ER

Anode basket

Steel cathode

Salt (12”)
LiCl-KCl-PuCl₃
(FP&TRU)

Cadmium pool (6”)

Pure U
(Dendritic)

Dual Cathodes ER

CRITICALITY ANALYSIS OF ELECTROREFINER

Vector of U and Pu in the recycled nuclear fuel

<table>
<thead>
<tr>
<th>Fuel</th>
<th>U Vector</th>
<th>Pu Vector</th>
</tr>
</thead>
<tbody>
<tr>
<td>U/TRU</td>
<td>0.01% U-234</td>
<td>0.53% Pu-238</td>
</tr>
<tr>
<td></td>
<td>0.08% U-235</td>
<td>78.45% Pu-239</td>
</tr>
<tr>
<td></td>
<td>0.03% U-236</td>
<td>18.81% Pu-240</td>
</tr>
<tr>
<td></td>
<td>99.88% U-238</td>
<td>1.47% Pu-241</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.74% Pu-242</td>
</tr>
<tr>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>HALEU</td>
<td>0.01% U-234</td>
<td>0.93% Pu-238</td>
</tr>
<tr>
<td></td>
<td>5.81% U-235</td>
<td>88.83% Pu-239</td>
</tr>
<tr>
<td></td>
<td>2.66% U-236</td>
<td>9.74% Pu-240</td>
</tr>
<tr>
<td></td>
<td>91.52% U-238</td>
<td>0.47% Pu-241</td>
</tr>
<tr>
<td></td>
<td></td>
<td>0.04% Pu-242</td>
</tr>
<tr>
<td></td>
<td></td>
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</tr>
<tr>
<td>PWR</td>
<td>0.02% U-234</td>
<td>3.12% Pu-238</td>
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<tr>
<td></td>
<td>0.80% U-235</td>
<td>54.18% Pu-239</td>
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<tr>
<td></td>
<td>0.63% U-236</td>
<td>25.74% Pu-240</td>
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<tr>
<td></td>
<td>98.55% U-238</td>
<td>9.59% Pu-241</td>
</tr>
<tr>
<td></td>
<td></td>
<td>7.73% Pu-242</td>
</tr>
</tbody>
</table>
Multiplication factor as function of U mass in cathode

- $k_{\text{eff}}$ 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)
Multiplication factor as function of salt height in the tank

- $k_{\text{eff}}$ 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
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 $k_{\text{eff}} << 0.95$ 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 UF₆, 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.govSCALE/analysis/non-lwr-models-vol3
  - Vol. 5 models will be added soon
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.
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**

---

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

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

(A1) $\text{Na}(l) + \text{H}_2\text{O}(l) \rightarrow \text{NaOH}(a) + \frac{1}{2}\text{H}_2$

(A2) $2\text{Na}(g,l) + \text{H}_2\text{O}(g,l) \rightarrow \text{Na}_2\text{O}(a) + \text{H}_2$

(A3) $2\text{Na}(g,l,a) + \frac{1}{2}\text{O}_2$ or $\text{O}_2 \rightarrow \text{Na}_2\text{O}(a)$ or $\text{Na}_2\text{O}_2(a)$

(A4) $\text{Na}_2\text{O}_2(a) + 2\text{Na}(g,l) \rightarrow 2\text{Na}_2\text{O}(a)$

(A5)
- $\text{Na}_2\text{O}(a) + \text{H}_2\text{O}(g,l) \rightarrow 2\text{NaOH}(a)$
- $\text{Na}_2\text{O}_2(a) + \text{H}_2\text{O}(g,l) \rightarrow 2\text{NaOH}(a) + 0.5\text{O}_2$

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\textsubscript{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 m³

HEPA-filtered ventilation system
- 2X air exchanges per hour (assumed)
- Maintains -2” H₂O 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

![Graph showing temperature and time relationship for various levels in a cask.]

- Clad melting
- Fuel melting
- Level 10
- Level 9
- Level 8
- Level 7
- Level 6
- Level 5
- Level 4
- Level 3
- Level 2
- Level 1
- Debris reflector
- Debris Inlet
- FUM bottom

- Start of fuel candling and collapse
- Debris relocation to the bottom of the fuel cask
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.

- Aerosol mass median dia 0.15 to 0.4 µm at HEPA inlet.
- Debris relocation to the bottom of the fuel cask.
- Captured on filters.
- Environment airborne + settled.
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
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” H₂O 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
Sodium purification system pipe break scenario

- Sodium fires generate lots of aerosols
  - $2 \text{Na} + \frac{1}{2} \text{O}_2 \rightarrow \text{Na}_2\text{O}$ (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
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

[Graphs showing mass burned and mass spilled over time for different spray scenarios.]
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

Reprocessing and fuel fabrication accident analysis

Example of a fire scenario

Example of an explosion scenario

- Pressure response figure below shows immediate failure of HEPA Filter 7 at the exit of the PPC.
- The dissipation of the pressure from the explosion also fails the final exhaust filter within 13 seconds.

Pressure response at the filters between the PPC and the stack:
- HEPA Filter 1 fails
- HEPA Filter 4 fails
- AFS & VFS always <0 psig

2.49 kPa = HEPA overpressure failure

Activity distribution in the first hour after the accident:
- Activity distribution above shows a large release to the environment due to the failure of the two HEPA filters between the PPC and the plant stack.
- Pre-filter 1 remains intact and retains show larger aerosols.

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

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
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

CDA no mixing release insights

Halogen
- 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_{2}O_{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

No mixture assumption
IAEA –TECDOC-2006 Notes

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