Outline

NRC strategy for non-LWR source term analysis

Project scope

Overview of Fluoride-salt-cooled High-temperature Reactor (FHR)

FHR reactor fission product inventory/decay heat methods & results

MELCOR molten salt models

FHR plant model and source term analysis

Summary

Background slides

• SCALE

• MELCOR
Integrated Action Plan (IAP) for Advanced Reactors

Near-Term Implementation Action Plan

Strategy 1
Knowledge, Skills, and Capacity

Strategy 2
Analytical Tools

Strategy 3
Flexible Review Process

Strategy 4
Industry Codes and Standards

Strategy 5
Technology Inclusive Issues

Strategy 6
Communication

ML17165A069
These Volumes outline the specific analytical tools to enable independent analysis of non-LWRs, “gaps” in code capabilities and data, V&V needs and code development tasks.
NRC strategy for non-LWR analysis (Volume 3)

Evaluation Model and Suite of Codes

Code strategy for source term
Role of NRC severe accident codes

**RES**
- SCALE (w/ Sensitivity/Uncertainty Quantification)
  - Isotopic Inventories, Decay Heat, Kinetics and Power Distribution Parameters
- MELCOR (w/ Sensitivity/Uncertainty Quantification)
  - Radionuclide Source Term
- MACCS, RADTRAD, RASCAL, etc. (w/ Sensitivity/Uncertainty Quantification)
  - Dose, Health Effects, Economic/Societal Consequences

**NMSS**
- Storage & Transport of Materials
- Material Processing for Applicable Designs

**NRR**

**Safety Review (Regulations)**
- Siting and Safety Analysis\(^1\,^2\) - 10 CFR 100.21, 10 CFR 50.34, Part 52 (various)
- Control Room Habitability\(^3\) - 10 CFR 50, Appendix A, GDC-19
- Technical Support Center Habitability\(^4\) - 10 CFR 50, Appendix E, 10 CFR 50.47
- Severe Accidents, FSAR Chapter 19
- Emergency Planning – 10 CFR 50.160 (expected future use assuming this regulation is promulgated)
- Emergency Response – Nuclear/Radiological Incident Annex (NRIA) to the National Response Framework (NRF)

**Environmental Review (Regulations)**
- Environmental Report - 10 CFR 51.50
- Draft EIS (general) - 10 CFR 51.70
- Draft EIS (CP, ESP, COL) - 10 CFR 51.75
- Severe Accident Mitigation Design Alternatives (SAMDA) - 10 CFR 51.55

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Dose Criteria Reference Values (10 CFR 50/52)

1. An individual located at any point on the boundary of the exclusion area for any 2-hour period following the onset of postulated fission product release, would not receive a radiation dose in excess of 0.25 Sv (25 rem) total effective dose equivalent (TEDE).
2. An individual located at any point on the outer boundary of the low population zone, who is exposed to the radioactive cloud resulting from the postulated fission product release (during the entire period of its passage), would not receive a radiation dose in excess of 0.25 Sv (25 rem) TEDE.
3. Adequate radiation protection is provided to permit access to and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 0.05 Sv (5 rem) TEDE.
4. Dose criteria not in regulation but found in NUREG-0737/NUREG-0696. GDCs are applicable to light-water reactors. Non-LWRs will have principal design criteria (PDCs) which may have a similar requirement.
Project Scope
Project objectives

Understand severe accident behavior
  • Provide insights for regulatory guidance

Facilitate dialogue on staff’s approach for source term

Demonstrate use of SCALE and MELCOR
  • Identify accident characteristics and uncertainties affecting source term
  • Develop publicly available input models for representative designs
Project scope

Full-plant models for three representative non-LWRs (FY21)

- Heat pipe reactor – INL Design A
- Pebble-bed gas-cooled reactor – PBMR-400
- Pebble-bed molten-salt-cooled – UCB Mark 1

FY22

- Molten-salt-fueled reactor – MSRE
- Sodium-cooled fast reactor – To be determined
Project approach

1. Build MELCOR full-plant input model
   • Use SCALE to provide decay heat and core radionuclide inventory

2. Scenario selection

3. Perform simulations for the selected scenario and debug
   • Base case
   • Sensitivity cases
Advanced Reactor Designs

- **Liquid Metal Cooled Fast Reactors (LMFR)**
  - TerraPower/GEH (Natrum)*
  - GEH PRISM (VTR)
  - Oklo
  - Advanced Reactor Concepts
  - Sodium-Cooled

- **High-Temperature Gas-Cooled Reactors (HTGR)**
  - X-energy*
  - Framatome
  - StarCore
  - MIT

- **Molten Salt Reactors (MSR)**
  - Kairos*
  - Kairos (Hermes|RTR)
  - Liquid Salt Cooled

- **Micro Reactors**
  - Westinghouse (eVinci)
  - BWX Technologies
  - X-energy
  - Radiant | RTR
  - Transportable

**TRISO Fuel**

- General Atomics (EM2)
- General Atomics

**Liquid Salt Fueled**

- Terrestrial*
- TerraPower
- Southern (TP MCFR) | RTR
- ACU | RTR *
- Elysium
- Thorcon
- Muons
- Flibe
- Alpha Tech
- Liquid Salt Fueled

**LEGEND**

- ARDP Awardees
- Demo Reactors
- Risk Reduction
- * Preapplication
- RTR Research/Test Reactor
- In Licensing Review

**Micro Reactors**

- Oklo
- Stationary

**Ultra Safe | RTR**
Fluoride-Salt-Cooled High-Temperature Reactor (FHR)
Molten-salt reactors (1/3)

Aircraft Nuclear Propulsion Program (ANP) – 1946-1961

- Long-term strategic bomber operation using nuclear power
- ORNL developed the nuclear concept with the Aircraft Reactor Experiment (ARE)
  - Originally sodium cooled, but shifted to molten salt
  - 2.5 MW molten salt-cooled reactor operated for 96-MW-hours in November 1954
- Three Heat Transfer Reactor Experiments at Idaho National Laboratory to demonstrate the jet engine propulsion
- Aircraft Shield Test (AFT) – B-36 with an operating reactor flew 47 times over West Texas and New Mexico to study shielding (i.e., the reactor was operating but not part of the propulsion system)
- Terminated due to inventing ballistic missile and supersonic aviation
Molten-salt reactors (2/3)

ORNL Molten Salt Reactor (MSR)
- AEC funded the Molten Salt Reactor Experiment (MSRE)
- Operated from 1965 to 1969
- 30 MWt
- Coolant was FLiBe molten salt
- Fuel was dissolved in coolant (molten fuel)
Molten-salt reactors (3/3)

UCB Mark 1 – circa 2013

- Coolant is FLiBe molten salt
- Core is TRISO fuel in a pebble-bed geometry
- Design description

- Used for the SCALE/MELCOR demonstration project
Reactor
- 236 MW\(_{th}\) / 100 MW\(_{e}\)
- Atmospheric pressure
- 600°C core inlet
- 700°C core outlet
- 976 kg/s core flowrate
- FLiBe molten salt coolant

Core
- 470,000 fueled pebbles + 218,000 unfueled pebbles in core and defueling chute
- 180 MWd/kgHM discharge burnup
- 19.9% enrichment
- Online refueling

Secondary system: gas-turbine at 18.6 bar with natural gas co-firing capability
Recirculation loops
- Salt pumps in the hot well with FLiBe free surface
- 2X cross-over legs to coiled tube air heaters (CTAH)
- 2X cold legs with standpipes with free surface
- Drain tank with freeze valve
Containment

• Most reactor and secondary components below-grade
• Compartmentalized building
• Low-free-volume reactor cavity with fire-brick insulation, steel liner, and concrete walls
• Shield building (above grade)
UCB Mark 1 (4/4)

Direct Reactor Auxiliary Cooling System (DRACS)

- 3 trains – 2.36 MW/train
  - 236 MWt reactor
- Each train has 4 loops in series
  - Primary coolant circulates to DRACS heat exchanger
  - Molten-salt loop circulates to the thermosyphon-cooled heat exchangers (TCHX)
  - Water circulates adjacent to the secondary salt tube loop in the TCHX
  - Natural circulation air circuit cools and condenses steam
- Start-up: Reactor coolant pump trip causes ball in valve to drop

Reactor cavity cooling subsystem (RCCS) surrounds reactor cavity

- Thermal protection of the concrete
UCB Mark 1 fuel

TRISO particle
- TRISO is a portmanteau of tristructural isotropic
- Kernel – 1.5 g of UCO, 200 µm radius
- Porous carbon buffer layer
- 3 coatings to contain fission products

TRISO pebble
- Contains 4730 TRISO particles
- 30 mm diameter
- 1 mm graphite outer shell
- TRISO particles are distributed in the carbon matrix region between the solid core and outer shell
Fluoride-salt-cooled High-Temperature Reactor Fission Product Inventory/Decay Heat Methods and Results
FHR analysis with SCALE

• Objective
  • Provide input for MELCOR accident simulation
    ▪ Radionuclide inventory
    ▪ Decay heat profile
    ▪ Reactivity feedback coefficients
    ▪ Reactivity from xenon transient

• Approach
  • Apply SCALE to generate fuel composition for an equilibrium core
  • Equilibrium core – operated for several years so the average burnups are no longer changing
  • Evaluate neutronic characteristics
**SCALE capabilities used:**

- **Codes:**
  - ORIGEN for depletion
  - KENO-VI 3D Monte Carlo neutron transport

- **Data:** ENDF/B-VII.1 nuclear data library*

**Sequences:**

- CSAS for criticality/reactivity
- TRITON for reactor physics & depletion

*A NUREG about Nuclear Data Assessment for Advanced Reactors summarizing the outcome of a recently concluded NRC-sponsored project is going to be published soon.*
Relevant characteristics and differences to High Temperature Gas-cooled Reactors:

- **Fuel:**
  - UCO fuel in TRISO particles in fuel pebbles
  - TRISO particles located in shell instead of sphere
- **Coolant:** FLiBe salt instead of helium
- **Moderator:** graphite
Neutronics overview (2/2)

- Challenges for modeling:
  - Tritium production in FLiBe
  - TRISO particles with very high packing fraction in shell
  - Fuel pebble inlet and outlet geometry
  - Fuel and unfueled/graphite pebbles in different zones of the core

- Validation
  - SCALE validation with HTGR experiments partially applicable

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## UCB Mark 1 Model Description

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor power</td>
<td>236 MWth</td>
</tr>
<tr>
<td>UCO fuel density</td>
<td>10.5 g/cc</td>
</tr>
<tr>
<td>Uranium enrichment</td>
<td>19.9 wt.%</td>
</tr>
<tr>
<td>Fuel kernel radius</td>
<td>0.2 mm</td>
</tr>
<tr>
<td>Particle coating layer materials (starting from kernel)</td>
<td>Buffer/PyC/SiC/PyC</td>
</tr>
<tr>
<td>Fuel particle coating layer thickness</td>
<td>0.100/0.035/0.035/0.035 mm</td>
</tr>
<tr>
<td>Number of particles in pebble</td>
<td>4,730</td>
</tr>
<tr>
<td>Particle packing fraction in fuel pebble</td>
<td>40%</td>
</tr>
<tr>
<td>Radius of fuel pebble</td>
<td>1.5 cm</td>
</tr>
<tr>
<td>Inner/outer radius of fuel zone</td>
<td>1.25/1.40 cm</td>
</tr>
<tr>
<td>Number of fuel pebbles</td>
<td>470,000</td>
</tr>
<tr>
<td>Number of unfueled/graphite pebbles</td>
<td>218,000</td>
</tr>
<tr>
<td>Pebble packing fraction</td>
<td>60%</td>
</tr>
<tr>
<td>Core Inner reflector radius</td>
<td>35 cm</td>
</tr>
<tr>
<td>Outer fuel pebble region radius</td>
<td>105 cm</td>
</tr>
<tr>
<td>Outer graphite pebble region radius</td>
<td>125 cm</td>
</tr>
<tr>
<td>Volume of active fuel region</td>
<td>10.4 m³</td>
</tr>
<tr>
<td>Average pebble thermal power</td>
<td>500 W</td>
</tr>
<tr>
<td>Average pebble discharge burnup</td>
<td>180 GWd/MTIHM</td>
</tr>
<tr>
<td>Average pebble full-power lifetime</td>
<td>1.40 years</td>
</tr>
</tbody>
</table>

**SCALE model developed based on:**


Analysis areas

1. Verification of multigroup physics
2. Generation of equilibrium core
3. Power profile and neutron spectrum
4. Temperature feedback
5. Decay heat
6. 1-group cross sections
7. Tritium production
8. Xenon reactivity
1. Verification of multigroup physics for UCB Mark 1

- Comparison of multigroup (MG) calculation with continuous energy (CE) calculations for a pebble depletion problem

- **Why not always run CE?**
  - Significant modeling time: random distributions or particle arrays without permitting particle clipping
  - Significant computation time: many cells/surfaces (consider thousands of particles) and use of CE data

- **SCALE’s MG approach for double-heterogeneous systems:**
  - Two self-shielding calculations: (1) particle in graphite matrix, (2) pebble in lattice of pebbles
  - Generation of problem-dependent cross sections for the fuel region through user-friendly input block
  - The MG calculation is 5 times faster than the CE lattice calculation, and 24 times faster than the CE calculation with a random particle distribution

<table>
<thead>
<tr>
<th></th>
<th>CE, random</th>
<th>CE, lattice</th>
<th>MG</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Calculation:</strong></td>
<td>~79 minutes</td>
<td>15.28 minutes</td>
<td>3.25 minutes</td>
</tr>
<tr>
<td></td>
<td>×4.7</td>
<td>×24</td>
<td></td>
</tr>
</tbody>
</table>
1. Single pebble models

1. CE model: Random particle distribution
2. CE model: particle lattice (no clipping)
3. CE model: particle lattice (clipping)
4. MG model

Note:
• CE-random results are average of 10 realizations
• All models contain the same amount of fuel
1. Single pebble initial criticality

<table>
<thead>
<tr>
<th>Model</th>
<th>CZP</th>
<th>HFP</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>$k_{\text{eff}}$</td>
<td>$\Delta \rho$ [pcm]</td>
</tr>
<tr>
<td>CE, random</td>
<td>1.52539</td>
<td>(ref)</td>
</tr>
<tr>
<td>no clipping</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CE, lattice</td>
<td>1.52449</td>
<td>-39</td>
</tr>
<tr>
<td>no clipping</td>
<td></td>
<td></td>
</tr>
<tr>
<td>CE, lattice</td>
<td>1.51939</td>
<td>-259</td>
</tr>
<tr>
<td>clipping</td>
<td></td>
<td></td>
</tr>
<tr>
<td>MG</td>
<td>1.51986</td>
<td>-239</td>
</tr>
</tbody>
</table>

**Result:** MG $k_{\text{eff}}$ calculations show good agreement with reference CE result independent of the temperature.

- **CZP:** all materials 300K
- **HFP:** Fuel 1003K, TRISO layers 973K, graphite center 983K, outer graphite shell 957K, coolant 923K
- All statistical errors of the Monte Carlo calculations < 20 pcm
1. Single pebble $k_{\text{eff}}$ over the course of depletion

Result:
MG bias remains below 260 pcm over depletion

Calculation details:
• TRITON-KENO depletion of the HFP case
• 540.54 days at 333 MW/MTIHM
1. Single pebble nuclide density comparison over depletion

Comparison of MG against CE random:

**Result:** MG bias remains below 3% for relevant nuclide densities over depletion
1. MG performance summary for UCB Mark 1

- We confirmed the performance of SCALE’s MG capability for double-heterogeneous systems in terms of $k_{\text{eff}}$ and nuclide densities in a UCB Mark 1 single pebble depletion calculation.
- SCALE’s MG capability permits the calculation of accurate results in a much-reduced runtime (factor of 24 when compared to reference CE calculations).
2. Generation of equilibrium full core

**Goal:** Determine fuel composition of pebbles in a full core corresponding to an equilibrium state

**Boundary conditions:**
- Pebble final discharge burnup: 180 GWD/tHM
- Average number of passes per pebble: 8
- Average power: 333 MW/tHM
- Rods fully withdrawn

**Full core model discretization:**
- 10 axial zones of equal volume
- 3 radial zones with 1/8<sup>th</sup>, 6/8<sup>th</sup>, 1/8<sup>th</sup> fractional volumes

**Assumptions:**
- All pebbles within a zone contain the same fuel composition
- Fuel composition within a zone represents average of individual pebbles of different passes/burnups in this zone
2. Generation of isotopics for an equilibrium state

Fuel pebble burnup (GWd/MTIHM) in each axial zone depending on the pass through the core assuming constant axial/radial power:

<table>
<thead>
<tr>
<th>axial zone</th>
<th>pass through the core</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>1</td>
</tr>
<tr>
<td>10</td>
<td>21.4</td>
</tr>
<tr>
<td>9</td>
<td>19.1</td>
</tr>
<tr>
<td>8</td>
<td>16.9</td>
</tr>
<tr>
<td>7</td>
<td>14.6</td>
</tr>
<tr>
<td>6</td>
<td>12.4</td>
</tr>
<tr>
<td>5</td>
<td>10.1</td>
</tr>
<tr>
<td>4</td>
<td>7.9</td>
</tr>
<tr>
<td>3</td>
<td>5.6</td>
</tr>
<tr>
<td>2</td>
<td>3.4</td>
</tr>
<tr>
<td>1</td>
<td>1.1</td>
</tr>
</tbody>
</table>

Re-enter pebble into core to complete 8 passes total

Mix fuel compositions of these burnups to get average composition of axial zone 3

Pebbles of the various passes:

1 2 3 4 5 6 7 8 9 10
2. Approach to generate equilibrium inventory

1. Depletion of surrogate pebbles in a **core slice model** to capture average spectral effects in equilibrium environment

2. Depletion of every pebble according to its detailed **power** and spectral history (pass and zone in 3D core) based on average conditions from slice depletion

3. Reconstruction of 3D core equilibrium composition according to axial/radial zones

4. Check convergence for keff and core-average fuel composition: stop or return to step 1 with new core-average fuel composition

**Outer iteration:**
1. Constant power
2. 3D power map
2. Slice depletion model

Why a slice and not a single pebble:
• Representative moderator/fuel ratio
• Representative neighboring conditions (spectral effects)

Depletion model:
• Slice through center of the core
• Depletion of surrogate pebbles surrounded by core-average fuel composition
• Axially reflected, radially vacuum boundary conditions
2. $K_{\text{eff}}$ and nuclide density convergence

- Outer iteration 1: convergence of $k_{\text{eff}}$ and nuclide densities achieved after 8 inner iterations
- Outer iteration 2 using 3D power map showed similar convergence behavior
3. Full core power profile

Results:
1. Power peak in the lower core region in eq. core due to increasing burnup with axial height
2. Difference between power profiles of the two outer iterations very small with max. 6% in the lowermost zone
3. Example fuel cell flux spectrum comparison

- UCB Mark 1 and PBMR show a larger thermal peak compared to LWR.
- UCB Mark 1 shows smaller fast flux due to scattering with the salt.
3. Energy-dependent flux profile
3. 3D full core flux visualizations

Fast flux, $E > 0.625 \text{ eV}$

Thermal flux, $E < 0.625 \text{ eV}$

Total flux at the axial center of the core
3. Radial flux distribution at axial core center (axial zone 5)
3. Axial flux distribution in the fuel region

![Diagram showing axial flux distribution]

- Fast flux (red)
- Thermal flux (blue)

**Normalized Flux** vs. **Axial Location**

Normalized flux is plotted against axial location in centimeters. The graph compares fast and thermal flux distributions within the fuel region.
4. Reactivity coefficients

- Isothermal temperature coefficient calculation:
  - $k_{\text{eff}}$ calculations with material temperatures varying over a range of several hundred K
  - Assuming constant temperature within material
  - Fitting of reactivity $\rho$ to determine coefficient
- $\beta_{\text{eff}}$ and coolant void coefficient

### Nominal temperatures:
- Fuel: 1003 K
- Salt coolant: 923 K
- Graphite moderator*: 973/983 K
- Inner graphite reflector: 873 K
- Outer graphite reflector: 973 K

*All carbonaceous materials in fuel pebbles

### Reactivity coefficients

<table>
<thead>
<tr>
<th>Component</th>
<th>Temperature Reactivity Coefficient at nominal temperature [pcm/K]</th>
</tr>
</thead>
<tbody>
<tr>
<td>Salt coolant</td>
<td>-0.48</td>
</tr>
<tr>
<td>Fuel</td>
<td>-3.90</td>
</tr>
<tr>
<td>Graphite moderator</td>
<td>-1.10</td>
</tr>
<tr>
<td>Inner graphite reflector</td>
<td>+1.21</td>
</tr>
<tr>
<td>Outer graphite reflector</td>
<td>+0.61</td>
</tr>
</tbody>
</table>

Component | Value [pcm]  
- $\beta_{\text{eff}}$ | 541 ± 20  
- Coolant void | -5094 ± 21
4. Isothermal temperature coefficients

1. Linear fit for salt temperature coefficient
2. Polynomial fit or tabulated values for fuel, moderator, and graphite temperature coefficients

\[
\rho = a + bT + cT^2 + dT^3
\]

<table>
<thead>
<tr>
<th></th>
<th>a</th>
<th>b</th>
<th>c</th>
<th>d</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel</td>
<td>4.57E-02</td>
<td>-7.08E-05</td>
<td>1.59E-08</td>
<td></td>
</tr>
<tr>
<td>Moderator</td>
<td>-2.02E-03</td>
<td>-2.48E-05</td>
<td>3.88E-08</td>
<td>-2.16E-11</td>
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<tr>
<td>Inner graphite</td>
<td>-2.18E-02</td>
<td>2.07E-05</td>
<td>-7.55E-09</td>
<td></td>
</tr>
<tr>
<td>Outer graphite</td>
<td>-3.10E-02</td>
<td>3.49E-05</td>
<td>-1.31E-08</td>
<td></td>
</tr>
</tbody>
</table>

2\sigma statistical error bars are displayed
5. Generation of decay heat file for MELCOR

Fuel composition files for the 8 passes for all 30 zones of the core from generation of equilibrium core

Average compositions together according to zone volumes in the core to obtain core-average fuel composition

10-day decay calculation with ORIGEN, generating new composition file

Generation of core-average inventory JSON file using ORIGEN composition file

Conversion of JSON file to MELCOR DCH file while scaling to actual initial heavy metal mass in the core (0.705 tHM)
5. Generation of decay heat file for MELCOR

Relative contribution of top fission products

Relative contribution of top actinides
5. Decay heat comparisons

- UCB Mark 1: equilibrium core
- PWR: approximate end of cycle core (mixture of assemblies at burnup of 20, 40, 60 GWd/tHM)
6. Towards rapid inventory calculations with ORIGAMI

**Purpose of 1-group cross section analysis:** understand the spectral variations and their impact on 1-group cross sections which influence all inventory calculations

- Only small variation of 1-group removal cross section over depletion
- Small changes visible mainly in Pu-240
6. Axial variation of 1-group removal cross section

Axial variation:
• Low variation within main core region
• Significant variation in inlet/outlet regions
• Opposing trends for certain nuclides, such as $^{239}$Pu vs. $^{240}$Pu
6. Radial variation of 1-group removal cross section

Radial variation:
Significant radial variation for various nuclides
7. Tritium production

Tritium overview

- FHR uses FLiBe coolant
  - Lithium is enriched to >99.5% Li-7 because Li-6 is a neutron poison
- Li-6 and Li-7 react with neutrons to produce tritium
  - $^6\text{Li} + n \rightarrow ^4\text{He} + ^3\text{H}$
  - $^7\text{Li} + n \rightarrow ^4\text{He} + ^3\text{H} + n'$
- Tritium is a potential radiological dose hazard

- Mass of FLiBe defined in the ORIGEN model is the total FLiBe mass in the entire system
  - To irradiate just the FLiBe in the core at a given time, we scale the flux in our ORIGEN model based on what volume fraction of FLiBe is in the core

- ORIGEN flux is equal to $\phi \times \left( \frac{V_{\text{core}}}{V_{\text{total}}} \right)$

TRITON

- Determine the flux spectrum and 1-group cross sections in FLiBe in this core

ORIGEN

- Irradiate FLiBe using explicit flux magnitude scaled based on the fraction of system FLiBe in the core at any given time
7. Equilibrium tritium production rate

- SCALE-predicted equilibrium value is 0.021 mol/day
  - Equilibrium value from Cisneros was 0.023 mol/day

- Equilibrium is a balance between Li-6 production and destruction
  - $^9\text{Be} + n \rightarrow ^4\text{He} + ^6\text{Li} + e^- + \bar{\nu}_e$
  - $^6\text{Li} + n \rightarrow ^4\text{He} + ^3\text{H}$

- The calculated behavior is consistent with established trends in the literature

$^3\text{H} t_{1/2} = 12.32$ years

0.021 mol/day

Net Tritium Production Rate (mol/day)

Time (Years)
7. Sensitivity analysis on tritium production

• We ran 5,000 combinations of initial Li-7 enrichment and flux using SAMPLER to determine their impact on equilibrium tritium production

• Variations in initial tritium production rate are quite large and depend on flux and initial Li-7 enrichment
  • Li-6 is a neutron poison, so FHR systems seek to enrich coolant in Li-7
  • Natural Li is 7.59% Li-6

<table>
<thead>
<tr>
<th>Property</th>
<th>Minimum Value</th>
<th>Maximum Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Flux (n/cm²-s)</td>
<td>$3.528 \times 10^{14}$</td>
<td>$4.312 \times 10^{14}$</td>
</tr>
<tr>
<td>Initial Li-7 Enrichment (w/o)</td>
<td>99.95</td>
<td>100.0</td>
</tr>
</tbody>
</table>
7. Sensitivity analysis on tritium production

Initial Li-7 enrichment has no effect on equilibrium tritium production rate, while flux has a significant impact.

- No correlation for initial Li-7 enrichment
- Strong correlation for neutron flux
8. Transient xenon reactivity

- Steady-state Xe-135 reactivity worth is -6.48$
- Using equilibrium I-135 and Xe-135 concentrations from UCB Mark 1 model, we can calculate time-dependent concentrations analytically
- When flux goes to zero, Xe-135 inventory is dictated only by decay of I-135 and Xe-135
- Peak Xe-135 reactivity is -18.6$ and occurs at 9.49 hours
- Xe-135 reactivity drops below steady-state value after 34.67 hours

\[ \frac{dI}{dt} = \gamma_I \Sigma_f \phi - \lambda_I I \]
\[ \frac{dX}{dt} = \gamma_X \Sigma_f \phi + \lambda_I I - \lambda_X X - \sigma_a X \phi \]
Demonstrated SCALE’s capabilities for FHR modeling

- SCALE’s multigroup physics was confirmed adequate through FHR fuel pebble analysis: $k_{\text{eff}}$ bias smaller than 260 pcm, while achieving 24 times faster runtime
- Fuel compositions for an equilibrium core were developed using an iterating scheme
- Power profiles and decay heat were determined for equilibrium core
- Temperature feedback: linear behavior found for salt, nonlinear trend for fuel and for materials containing graphite
- Strong radial variation for 1-group cross section was observed, while axial variation was limited to inlet/outlet regions
- Tritium production rate in coolant salt was estimated
- Preliminary results for time-dependent Xe-135 concentration
Added molten salt as working fluid

Fission product release
• Release from TRISO kernel
• Radionuclide distributions within the layers in the TRISO particle and compact
• Liquid-phase fission product chemistry and transport model

Additional core models
• Graphite oxidation
• Intercell and intracell conduction
• Convection & flow

Fluid point kinetics (liquid-fueled molten salt reactors)
Stage 0: Normal Operation
Establish thermal state

Time constant in FHR graphite structures is very large
Reduce heat capacities for structures to reach steady state thermal conditions.
Reset heat capacities after steady state is achieved.

Stage 1: Normal Operation
Diffusion Calculation

Establish steady state distribution of radionuclides in TRISO particles, and matrix

Stage 2: Normal Operation
Transport Calculation

Calculate steady state distribution of radionuclides into the molten salt (formation of soluble, colloidal fission products, deposition on surfaces, convection through flow paths)

Stage 3: Accident
Diffusion & Transport calculation

Calculate accident progression and radionuclide release

Cesium Release from the Pebbles

Fraction of initial inventory (–)

Time (sec)

Power (MW)

Core Power
CTAH HX
UCB Reference

Cesium Release from the Pebbles

Fraction of initial inventory (–)

Time (sec)

Power (MW)

Core Reactivities
Fuel Temperature
Moderator
Outer Reflector
Core Reflector
Inner Reflector
Moderator
Reactor
Total Reactivity
Power

Cesium Release from the Pebbles

Fraction of initial inventory (–)

Time (h)
Core components

• Pebble Bed Reactor Fuel/Matrix Components
  ▪ Fueled part of pebble
  ▪ Unfueled shell (matrix) is modeled as separate component
  ▪ Fuel radial temperature profile for sphere

• Prismatic Modular Reactor Fuel/Matrix Components
  ▪ “Rod-like” geometry
  ▪ Part of hex block associated with a fuel channel is matrix component
  ▪ Fuel radial temperature profile for cylinder

Legend

- TRISO (FU)
- Fuel (FU)
- GRAPHITE
- Matrix (MX)
- Fluid B/C

Sub-component model for zonal diffusion of radionuclides through TRISO particle
Radionuclide Diffusion Release Model

Intact TRISO Particles
- One-dimensional finite volume diffusion equation solver for multiple zones (materials)
- Temperature-dependent diffusion coefficients (Arrhenius form)

\[
\frac{\partial C}{\partial t} = \frac{1}{r^n} \frac{\partial}{\partial r} \left( r^n D \frac{\partial C}{\partial r} \right) - \lambda C + \beta \\
D(T) = D_0 e^{-\frac{Q}{RT}}
\]

Data used in the demo calculation
[IAEA TECDOC-0978]

Diffusivity Data Availability

<table>
<thead>
<tr>
<th>Radionuclide</th>
<th>UO₂</th>
<th>UCO</th>
<th>PyC</th>
<th>Porous Carbon</th>
<th>SiC</th>
<th>Matrix Graphite</th>
<th>TRISO Overall</th>
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<tr>
<td>Ag</td>
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<td>Some</td>
<td>Extensive</td>
<td>Some</td>
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<td>I</td>
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<td>Not found</td>
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<tr>
<td>Sr</td>
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<td>Some</td>
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<td>Some</td>
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<td>Not found</td>
<td>Some</td>
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</table>

Data used in the demo calculation
[IAEA TECDOC-0978]

<table>
<thead>
<tr>
<th>Layer</th>
<th>Kr (m²/s)</th>
<th>O (J/mole)</th>
<th>Cs (m²/s)</th>
<th>O (J/mole)</th>
<th>Sr (m²/s)</th>
<th>O (J/mole)</th>
<th>Ag (m²/s)</th>
<th>O (J/mole)</th>
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<tr>
<td>Kernel (normal)</td>
<td>1.3E-12</td>
<td>126000.0</td>
<td>5.6-8</td>
<td>209000.0</td>
<td>2.2E-3</td>
<td>488000.0</td>
<td>6.75E-9</td>
<td>165000.0</td>
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<tr>
<td>Buffer</td>
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<td>0.0</td>
<td>1.0E-8</td>
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<td>1.0E-8</td>
<td>0.0</td>
<td>1.0E-8</td>
<td>0.0</td>
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<tr>
<td>PyC</td>
<td>2.9E-8</td>
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<td>6.3E-8</td>
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<tr>
<td>SiC</td>
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<td>303000.0</td>
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<td>268000.0</td>
<td>1.6E00</td>
<td>258000.0</td>
</tr>
</tbody>
</table>

Iodine assumed to behave like Kr
Radionuclide Release Models

- Recent failures – particles failing within latest time-step (burst release, diffusion release in time-step)
- Previous failures – particles failing on a previous time-step (time history of diffusion release)
- Contamination and recoil
Graphite Oxidation

Existing capability introduced with High-Temperature Gas-cooled Reactors (HTGRs)

Steam oxidation

\[ R_{OX,\text{steam}} = \frac{k_4 P_{H_2O}}{1 + k_5 P_{H_2}^{0.5} + k_6 P_{H_2O}} \]

Air oxidation

\[ R_{OX} = 1.7804 \times 10^4 \exp\left(-\frac{20129}{T}\right) \left(\frac{P}{0.21228 \times 10^5}\right)^{0.5} \]

Reactions

1. \( C + O_2 \rightarrow CO_2(g) \)
2. \( C + \frac{1}{2} O_2 \rightarrow CO(g) \)
3. \( CO(g) + \frac{1}{2} O_2(g) \rightarrow CO_2(g) \)
4. \( C + CO_2(g) \rightarrow 2CO(g) \)

\( R_{OX} \) is the rate term in the parabolic oxidation equation [1/s]
Effective conductivity prescription for pebble bed (bed conductance)

- Zehner-Schlunder-Bauer, without radiation heat transfer

\[ k_{\text{eff}} = (1 - \sqrt{1 - \varepsilon})k_f + (\sqrt{1 - \varepsilon})k_c(T, \varepsilon, k_f, k_s) \]

where:
- \( \varepsilon \) = Bed porosity [-]
- \( k_f \) = Fluid (FLiBe) conductivity [W/m/K]
- \( k_c \) = Effective bed conductivity [W/m/K], used with zero radiative conductivity
- \( k_s \) = Solid conductivity [W/m/K]
- \( T \) = Solid temperature [K]

- Effective fluid conductivity combines liquid and vapor contributions according to vapor fraction

- Radiative conductivity is combined by vapor fraction and used in ZSB model with radiation terms

\[ k_{\text{eff}} = \left(1 - \sqrt{1 - \varepsilon}\right)k_r + \left(1 - \sqrt{1 - \varepsilon}\right)k_f + \left(\sqrt{1 - \varepsilon}\right)k_c \left(T, D_p, \varepsilon, k_f, k_s, k_r \left(X_{\text{vapor}}\right)\right) \]

\[ k_r = 4\varepsilon\sigma T^3 D_p X_{\text{vapor}} \]
Heat transfer coefficient (Nusselt number) correlations for pebble bed convection:

- Isolated, spherical particles
- Use $T_{\text{film}}$ to evaluate non-dimensional numbers, use maximum of forced and free Nu

$$Nu_{\text{Free}} = 2.0 + 0.6 Gr_f^{1/4} Pr_f^{1/3} \quad Nu_{\text{Forced}} = 2.0 + 0.6 Re_f^{1/2} Pr_f^{1/3}$$

- Constants and exponents accessible by sensitivity coefficient

Flow resistance

- Packed bed pressure drop

$$K_L(\varepsilon, Re) = \left[ C_1 + C_2 \frac{1 - \varepsilon}{Re} + C_3 \left( \frac{1 - \varepsilon}{Re} \right)^C_4 \frac{(1 - \varepsilon)}{\varepsilon D_p} \right] (1 - \varepsilon)L$$

<table>
<thead>
<tr>
<th>Correlation</th>
<th>$C_1$</th>
<th>$C_2$</th>
<th>$C_3$</th>
<th>$C_4$</th>
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<tr>
<td>Ergun (original)</td>
<td>3.5</td>
<td>300.</td>
<td>0.0</td>
<td>-</td>
</tr>
<tr>
<td>Modified Ergun (smooth)</td>
<td>3.6</td>
<td>360.</td>
<td>0.0</td>
<td>-</td>
</tr>
<tr>
<td>Modified Ergun (rough)</td>
<td>8.0</td>
<td>360.</td>
<td>0.0</td>
<td>-</td>
</tr>
<tr>
<td>Achenbach</td>
<td>1.75</td>
<td>320.</td>
<td>20.0</td>
<td>0.4</td>
</tr>
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</table>
Point Kinetics Modeling

Standard treatment
\[ \frac{dP}{dt} = \left( \frac{\rho - \beta}{\Lambda} \right) P + \sum_{i=1}^{6} \lambda_i Y_i + S_0 \]

Feedback models
- User-specified external input
- FHR example includes multiple feedbacks
  - Fuel
  - Molten salt around the fuel
  - Inner reflector
  - Outer reflector and unfueled pebbles
  - Moderator (matrix around fueled pebbles)

\[ \frac{dY_i}{dt} = \left( \frac{\beta_i}{\Lambda} \right) P - \lambda_i C_i, \quad \text{for } i = 1 \ldots 6 \]
Point Kinetics Modeling (MSR)

Derived from standard PRKEs and solved similarly

\[
\frac{dP(t)}{dt} = \left( \frac{\rho(t) - \bar{\beta}(t)}{\Lambda} \right) P(t) + \sum_{i=1}^{6} \lambda_i C_i^C(t) + S_0
\]

\[
\frac{dC_i^C(t)}{dt} = \left( \frac{\beta_i}{\Lambda} \right) P(t) - \left( \lambda_i + \frac{2}{\tau_c} \right) C_i^C(t) + \left( \frac{V_c}{V} \right) \left( \lambda_i + \frac{2}{\tau_L} \right) C_i^L(t), \quad i = 1 \ldots 6
\]

\[
\frac{dC_i^L(t)}{dt} = \left( \frac{V_c}{\tau_c V_L} \right) C_i^C(t) - \left( \lambda_i + \frac{1}{\tau_L} \right) C_i^L(t), \quad i = 1 \ldots 6
\]

\[
\bar{\beta}(t) = \beta - \beta(t)_{lost} = \beta - \left( \frac{\Lambda}{P(t)} \right) \sum_{i=1}^{6} \lambda_i C_i^L(t)
\]

Feedback models

- User-specified external input
- Doppler
- Fuel and moderator density
- Flow reactivity feedback effects integrated into the equation set

Validated against MSRE zero-power flow experiments
Molten Salt Chemistry and Radionuclide Release

Radionuclides grouped into forms found in the Molten Salt Reactor Experiment

Model Scope

Evaluation of thermochemical state

- Gibbs Energy Minimization with Thermochimica
- Provides solubilities and vapor pressures

Thermodynamic database

- Generalized approach to utilize any thermodynamic database
- An example is the Molten Salt Thermal Database
  - FLiBe-based systems
  - Chloride-based systems

Solubility determined from empirical evidence (P. Britt ORNL 2017)

Solubilities mapped to 17 MELCOR fission product classes

Insoluble MELCOR classes are assigned to be colloidal
Fluoride-salt-cooled High-Temperature Reactor Plant Model and Source Term Analysis
Core and reactor vessel

Core nodalization – light blue lines
- Assumes azimuthal symmetry
- Subdivided into 11 axial levels and 8 radial rings
- Core cells model molten salt fluid volume, reflector structures, the pebble-bed core, and the pebbles in the defueling chute

Fluid flow nodalization – black boxes
- Molten salt enters through the downcomer and flows into the center reflector and into the bottom of the pebble bed
- Molten salt leaves through the periphery of the core and upwards through the refueling chute
- Unfueled graphite pebbles in box labeled “180”
Recirculation loops

Each loop has a pump, a heat exchanger, and a standpipe

Molten salt has free surface in the hotwell and the standpipes

Argon gas above the free surfaces with connection to the cover-gas system
  • Over-pressurization relief passes through the cover gas system
  • Cover gas enclosure leaks into the containment when over-pressurized

Secondary-side air cools primary-side molten salt
Direct Reactor Auxiliary Cooling System (DRACS)

3 trains – 2.36 MW/train
  • 236 MWt reactor

Each train has 4 loops in series
  • Primary coolant circulates to DRACS heat exchanger
  • Molten-salt loop circulates to the thermosyphon-cooled heat exchangers (TCHX)
  • Water circulates adjacent to the secondary salt tube loop in the TCHX
  • Natural circulation air circuit cools and condenses steam

Start-up: RCS-pump trip causes ball in valve to drop

Additional system information
  • DHXs are in the reactor vessel
  • TCHXs are in the shield building
Containment

Shield dome
- Protection against aircraft and natural gas detonations (co-fired turbine concept)
- Contains water for DRACS and RCCS
- DRACS air natural circulation chimneys connected to the shield dome

Reactor cavity
- Fire-brick insulation
- Low free volume
- Low-leakage bellows between reactor cavity and adjacent cavities

Separate compartments for the other RCS components
- Below-grade compartment includes the cover-gas enclosure for reactor cavity over-pressurization

Reactor cavity cooling subsystem in reactor cavity wall
- Water circulation
- Cooling tubes affixed to reactor cavity steel liner
- Cools concrete during normal operation

Leak rate assumed consistent with BWR Mark 1 reactor building
- 100% vol/day at 0.25 psig
MELCOR model inputs (1/2)

Equilibrium inventory and decay heat from SCALE
Radial and axial power profiles from SCALE
Reactivity feedbacks from SCALE
Cell-to-cell radial and axial heat transfer in the pebble bed and to adjacent reflector structures
  • Modified Zehner-Schlunder-Bauer model formulation
  • Combined conductive and radiative (when core uncovered) heat transfer depends on the coolant and fuel conductivities, fuel (graphite) emissivity, pebble bed porosity

Pebble bed friction losses – Achenbach pressure drop formulation
  • $K_{loss} = 2 + 320 \left( \frac{1 - \epsilon}{Re} \right) + 20 \left( \frac{1 - \epsilon}{Re} \right)^{0.4}$

Pebble to fluid heat transfer within a cell
  • Forced convection using Wakao correlation, $Nu = 2 + 1.1 Re^{0.66} Pr^{0.33}$
Fission product diffusivities through the TRISO and the pebble matrix from IAEA-TECDOC-978, Appendix A

- Primarily based on values from German experiments with UO₂ TRISO pebbles
  - UO₂ data can be easily updated to UCO data*
- Limited data based on nuclides of Xe, Cs, Sr, and Ag
- Iodine assumed to behave like Kr

*UCO TRISO thermal failure characteristics were not available, so UO₂ TRISO diffusivity and UO₂ failure data were used. Both are changeable through user input with design-specific data.
Scenarios

Three scenarios with a loss of secondary heat removal
• ATWS – Anticipated transient without SCRAM
• SBO – Station blackout
• LOCA – Loss-of-coolant accident

Sensitivity calculations included
• DRACS performance
• Alternate cover-gas system interconnections (LOCA only)
ATWS

Loss-of-on-site power with failure to SCRAM
  • Salt pumps shut off
  • Reactor fails to SCRAM
  • Secondary heat removal ends
  • 0 to 3 trains of DRACS operating

Includes preliminary analysis with xenon transient
  • Guided by ORNL calculations
  • Xenon reactivity feedback model being implemented into MELCOR
ATWS with 3xDRACS

Initial fuel heatup has strong negative fuel and moderator feedback that offsets positive reflector feedbacks

Strong negative xenon transient feedback *

3xDRACS exceeds core power after 330 s

* Xenon transient approximated.
ATWS with variable DRACS (semi-log)

Early power decrease to decay heat level is similar for all cases
- 1xDRACS and 2xDRACS cases exceed decay heat later

Fuel temperatures cool down according to DRACS heat removal rate
- 0xDRACS peak fuel temperature = 990 °C at 10^5 s (T_{sat} ~ 1350 °C )

* Xenon transient approximated.
ATWS with variable DRACS – (Linear scale)

When the total reactivity exceeds zero, the core power increases

• Increased power heats the fuel and reduces the positive fuel reactivity
• Core power eventually converges on the DRACS heat removal rate

The long-term fuel temperatures increase to offset changes in the xenon feedback

* Xenon transient approximated.
Station Blackout

Loss-of-onsite power with SCRAM

• Salt pumps shut off
• Reactor scrams
• Secondary heat removal ends
• Variable DRACS operating (percentage of 1xDRACS)

Unmitigated sensitivity case

• No DRACS and extended calculation to 7 days
SBO results (1/3)

DRACS cases illustrate degraded response
- Results for fraction of 1xDRACS
- >40% of one DRACS stops the temperature rise within 48 hr

DRACS power follows heat removal requirements
- 1xDRACS exceeds decay heat within 3 hr

![Graph showing Peak Fuel Temperature](image1)

![Graph showing Decay Heat and DRACS Heat Rejection](image2)
The TRISO failure fraction remains low ($1 \times 10^{-5}$) in the SBO with one DRACS operating.

- Higher TRISO failures were calculated as the DRACS degrades.

* UCO TRISO thermal failure characteristics were not available, so UO$_2$ TRISO diffusivity and UO$_2$ failure data were used. Both are changeable through user input with design-specific data.
SBO results (3/3)

The SBO with no DRACS was extended to 7 days

- No fuel uncoverage
- Peak fuel temperature approximately at Tsat (~1350 °C)
Loss-of-on-site power with LOCA

- Variable size leaks of the 3” pipe of the drain tank line
- Salt pumps shut off
- Reactor scrams
- Secondary heat removal ends
- 1 or no trains of DRACS operating
- With or without a cover gas connection path between the hotwell and the standpipes

Unmitigated sensitivity case

- No DRACS case extended to include fuel uncovery
LOCA results (1/6)

10% to 100% LOCA size did not significantly impact vessel boiloff timing

Cover gas connection (+ CG) between hotwell and standpipe prevents siphon
  • Stops initial drain down of vessel fluid
  • No significant impact on vessel boiloff timing
LOCA results (2/6)

Liquid drain down initially creates siphon and then low pressure region
- Causes a level difference between the core and downcomer

Core and downcomer levels equilibrate once there is gas flow around the loop
- Standpipe connections to the cover gas system are closed

10% LOCA at maximum point in the “siphon”

10% LOCA after equilibration
LOCA results (3/6)

LOCA cases without DRACS proceed to fuel uncovering at ~31 hr

Connection through the cover gas system keeps the DRACS active during the drain down

- Without the cover gas connection, the DRACS heat removal is delayed until the salt heats and expands

---

**Downcomer Level**

- No initial drain down with cover gas connection
- DRACS prevents boiloff

**Peak Fuel Temperature**

- Tsat at Core Outlet
- DRACS provides heat removal

---

100% LOCA cases

- 100% LOCA cases

---
We terminated the calculation at ~54 hr peak when the fuel kernel melting starts

- Reactor vessel wall and core barrel below the steel melting temperature
- Residual molten salt keeps the bottom level (level 1) at $T_{\text{sat}}$
- Upper vessel wall cools after downcomer salt level drops
- Pebbles and reflectors below graphite sublimation temperature (3600°C)
LOCA results (5/6)

TRISO failure rate extrapolated from available UO₂ TRISO data

- Correlation is based on data to 1800°C
- Initial failures set to $10^{-5}$ (0.001%)
- 0.017% of the TRISOs failed at 34 hr
- 7.5% of the TRISOs failed at 54 hr

Note:

“Fuel used thermal-physical properties of UO₂.
LOCA results (6/6)

Most of the fission product release from fuel is retained in the containment
  - Assumed hole size equivalent to 100% volume per day at 0.25 psig (8.7 in²)

The radionuclide distribution is affected by the timing of the release from the TRISO
  - Cesium release from the pebbles to the liquid molten salt starts earlier at lower fuel temperatures
  - Most aerosols leaving the primary system settle in the containment

Iodine Release and Distribution

Cesium Release and Distribution
Cesium vaporization from the molten salt

Molten salt chemistry and radionuclide release model calculates cesium and cesium fluoride release to the gas spaces

- Results use OECD/NEA JRC database for Thermochimica *
  - Includes vapor phase data for CsF

LOCA sequence

- No accelerated steady state
- No core uncovering through 24 hr
  - Cesium releases are from pebbles → liquid → gas

Model shows Cs/CsF vaporization to gas spaces at higher temperatures

* With modifications by Ontario Tech.
Conclusions

- Demonstrated use of SCALE and MELCOR for FHR safety analysis
- Simulated the entire accident starting with the initiating event
  - system thermal hydraulic response
  - fuel heat-up
  - heat transfer through the reactor to the surroundings
  - radiological release
- Evaluated effectiveness of passive mitigation features
Background Slides
Further SCALE analysis details
### Comparison of the FHR with other concepts

<table>
<thead>
<tr>
<th></th>
<th>Mk1 PB-FHR</th>
<th>ORNL 2012 AHTR</th>
<th>Westinghouse 4-loop PWR</th>
<th>PBMR</th>
<th>S-PRISM</th>
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<tr>
<td><strong>Coolant</strong></td>
<td>fibe</td>
<td>fibe</td>
<td>water</td>
<td>helium</td>
<td>sodium</td>
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<td><strong>Core inlet/outlet temperatures (°C)</strong></td>
<td>600-700</td>
<td>650-700</td>
<td>292/326</td>
<td>500/900</td>
<td>355/510</td>
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<tr>
<td><strong>Reactor thermal power (MWt)</strong></td>
<td>236</td>
<td>3400</td>
<td>3411</td>
<td>400</td>
<td>1000</td>
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<tr>
<td><strong>Reactor electrical power (MWe)</strong></td>
<td>100</td>
<td>1530</td>
<td>1092</td>
<td>175</td>
<td>380</td>
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<tr>
<td><strong>Fuel enrichment †</strong></td>
<td>19.90%</td>
<td>9.00%</td>
<td>4.50%</td>
<td>9.60%</td>
<td>8.93%</td>
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<tr>
<td><strong>Fuel discharge burn up (MWt-d/kg)</strong></td>
<td>180</td>
<td>71</td>
<td>48</td>
<td>92</td>
<td>106</td>
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<tr>
<td><strong>Fuel full-power residence time in core (yr)</strong></td>
<td>1.38</td>
<td>1.00</td>
<td>3.15</td>
<td>2.50</td>
<td>7.59</td>
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<tr>
<td><strong>Power conversion efficiency</strong></td>
<td>42.4%</td>
<td>45.0%</td>
<td>32.0%</td>
<td>43.8%</td>
<td>38.0%</td>
</tr>
<tr>
<td><strong>Core power density (MWt/m³)</strong></td>
<td>22.7</td>
<td>12.9</td>
<td>105.2</td>
<td>4.8</td>
<td>321.1</td>
</tr>
<tr>
<td><strong>Fuel average surface heat flux (MWt/m²)</strong></td>
<td>0.189</td>
<td>0.285</td>
<td>0.637</td>
<td>0.080</td>
<td>1.13</td>
</tr>
<tr>
<td><strong>Fuel specific surface area (area/volume) (1/m²)</strong></td>
<td>120</td>
<td>45</td>
<td>165</td>
<td>60</td>
<td>285</td>
</tr>
<tr>
<td><strong>Reactor vessel diameter (m)</strong></td>
<td>3.5</td>
<td>10.5</td>
<td>6.0</td>
<td>6.2</td>
<td>9.2</td>
</tr>
<tr>
<td><strong>Reactor vessel height (m)</strong></td>
<td>12.0</td>
<td>19.1</td>
<td>13.6</td>
<td>24.0</td>
<td>19.6</td>
</tr>
<tr>
<td><strong>Reactor vessel specific power (MWe/m³)</strong></td>
<td>0.866</td>
<td>0.925</td>
<td>2.839</td>
<td>0.242</td>
<td>0.292</td>
</tr>
<tr>
<td><strong>Start-up fissile inventory (kg-U235/MWe) ‡‡</strong></td>
<td>0.79</td>
<td>0.62</td>
<td>2.02</td>
<td>1.30</td>
<td>6.15</td>
</tr>
<tr>
<td><strong>EOC Cs-137 inventory in core (g/MWe) †</strong></td>
<td>30.8</td>
<td>26.1</td>
<td>104.8</td>
<td>53.8</td>
<td>269.5</td>
</tr>
<tr>
<td><strong>EOC Cs-137 inventory in core (Ci/MWe) †</strong></td>
<td>2672</td>
<td>2260</td>
<td>9083</td>
<td>4667</td>
<td>23359</td>
</tr>
<tr>
<td><strong>Spent fuel dry storage density (MWe-d/m³)</strong></td>
<td>4855</td>
<td>2120</td>
<td>15413</td>
<td>1922</td>
<td>-</td>
</tr>
<tr>
<td><strong>Natural uranium (MWe-d/kg-U)</strong> <strong>‡</strong></td>
<td>1.56</td>
<td>1.47</td>
<td>1.46</td>
<td>1.73</td>
<td>-</td>
</tr>
<tr>
<td><strong>Separative work (MWe-d/kg-SU)</strong> <strong>‡</strong></td>
<td>1.98</td>
<td>2.08</td>
<td>2.43</td>
<td>2.42</td>
<td>-</td>
</tr>
</tbody>
</table>

† For S-PRISM, effective enrichment is the Beginning of Cycle weight fraction of fissile Pu in fuel
‡‡ Assume start-up U-235 enrichment is 60% of equilibrium enrichment; for S-PRISM startup uses fissile Pu
* End of Cycle (EOC) life value (fixed fuel) or equilibrium value (pebble fuel)
** Assumes a uranium tails assay of 0.003

1. Single pebble nuclide density over depletion

CE random results:
1. Single pebble nuclide density comparison over depletion

Comparison of MG against CE random:

Result: MG bias remains below 3% for relevant nuclide densities over depletion
1. Single pebble nuclide density comparison of against reference CE random results
Monte Carlo calculation settings:

- 25,000 neutrons per cycle in 500 active and 100 inactive generations
- 1 node with 32 processors

### Single pebble runtime comparison

<table>
<thead>
<tr>
<th>Model</th>
<th>Runtime [min]</th>
</tr>
</thead>
<tbody>
<tr>
<td>CE, random no clipping</td>
<td>~79 (per realization)</td>
</tr>
<tr>
<td>CE, lattice no clipping</td>
<td>15.28</td>
</tr>
<tr>
<td>CE, lattice clipping</td>
<td>15.78</td>
</tr>
<tr>
<td>MG</td>
<td>3.25</td>
</tr>
</tbody>
</table>
2. Generation of isotopics for an equilibrium state

- **Outer iteration 1:**
  - Flat axial power profile
  - Consider only axial zones
  - No radial zones or radial power distribution
- **Outer iteration 2:**
  - Use axial and radial power profile from outer iteration 1
  - Consider axial and radial zones
  - Additional assumption: homogenization of compositions of all radial zones after each pass → initial composition for next pass
2. Convergence of results during iterations

Table 4. Slice depletion calculation iterations of outer iteration 1 (constant power).

<table>
<thead>
<tr>
<th>i</th>
<th>$t_{\text{final}}$ (days)</th>
<th>$b_{U_{\text{final}}}$ (GWD/THM)</th>
<th>$\frac{b_{U_{\text{final}}}}{180 \text{ GWD/THM}} - 1$</th>
<th>$k_{\text{eff}}$</th>
<th>$k_{i} - k_{i-1}$ (pcm)</th>
<th>$N_{D_{235U}}$ (atom/µ-cm)</th>
<th>$\frac{N_{D_{235U}}}{N_{D_{235U}-1}} - 1$</th>
<th>$N_{D_{239Pu}}$ (atom/µ-cm)</th>
<th>$\frac{N_{D_{239Pu}}}{N_{D_{239Pu}-1}} - 1$</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>540.54</td>
<td>144.56</td>
<td>-19.69%</td>
<td>1.32689</td>
<td>-</td>
<td>2.454E-03</td>
<td>-</td>
<td>2.063E-04</td>
<td>-</td>
</tr>
<tr>
<td>1</td>
<td>673.08</td>
<td>194.90</td>
<td>8.28%</td>
<td>1.03112</td>
<td>-29577</td>
<td>2.257E-03</td>
<td>-8.00%</td>
<td>2.147E-04</td>
<td>4.10%</td>
</tr>
<tr>
<td>2</td>
<td>621.63</td>
<td>187.44</td>
<td>4.13%</td>
<td>1.00464</td>
<td>-2648</td>
<td>2.273E-03</td>
<td>0.71%</td>
<td>2.104E-04</td>
<td>-2.02%</td>
</tr>
<tr>
<td>3</td>
<td>596.95</td>
<td>182.68</td>
<td>1.49%</td>
<td>1.00680</td>
<td>216</td>
<td>2.294E-03</td>
<td>0.91%</td>
<td>2.098E-04</td>
<td>-0.27%</td>
</tr>
<tr>
<td>4</td>
<td>588.21</td>
<td>180.39</td>
<td>0.22%</td>
<td>1.00954</td>
<td>274</td>
<td>2.304E-03</td>
<td>0.44%</td>
<td>2.102E-04</td>
<td>0.16%</td>
</tr>
<tr>
<td>5</td>
<td>586.94</td>
<td>180.15</td>
<td>0.08%</td>
<td>1.01046</td>
<td>92</td>
<td>2.307E-03</td>
<td>0.14%</td>
<td>2.114E-04</td>
<td>0.59%</td>
</tr>
<tr>
<td>6</td>
<td>586.45</td>
<td>180.01</td>
<td>0.00%</td>
<td>1.01120</td>
<td>74</td>
<td>2.314E-03</td>
<td>0.27%</td>
<td>2.123E-04</td>
<td>0.40%</td>
</tr>
<tr>
<td>7</td>
<td>586.43</td>
<td>179.84</td>
<td>-0.09%</td>
<td>1.01128</td>
<td>8</td>
<td>2.315E-03</td>
<td>0.04%</td>
<td>2.126E-04</td>
<td>0.18%</td>
</tr>
<tr>
<td>8</td>
<td>586.95</td>
<td>179.95</td>
<td>-0.03%</td>
<td>1.01126</td>
<td>-2</td>
<td>2.316E-03</td>
<td>0.06%</td>
<td>2.127E-04</td>
<td>0.05%</td>
</tr>
</tbody>
</table>

Table 5. Slice depletion calculation iterations using outer iteration 2 (axial/radial power profile).

<table>
<thead>
<tr>
<th>i</th>
<th>$t_{\text{final}}$ (days)</th>
<th>$b_{U_{\text{final}}}$ (GWD/THM)</th>
<th>$\frac{b_{U_{\text{final}}}}{180 \text{ GWD/THM}} - 1$</th>
<th>$k_{\text{eff}}$</th>
<th>$k_{i} - k_{i-1}$ (pcm)</th>
<th>$N_{D_{235U}}$ (atom/µ-cm)</th>
<th>$\frac{N_{D_{235U}}}{N_{D_{235U}-1}} - 1$</th>
<th>$N_{D_{239Pu}}$ (atom/µ-cm)</th>
<th>$\frac{N_{D_{239Pu}}}{N_{D_{239Pu}-1}} - 1$</th>
</tr>
</thead>
<tbody>
<tr>
<td>0</td>
<td>540.54</td>
<td>144.56</td>
<td>-19.69%</td>
<td>1.32689</td>
<td>-</td>
<td>2.454E-03</td>
<td>-</td>
<td>2.0881E-04</td>
<td>-</td>
</tr>
<tr>
<td>1</td>
<td>673.08</td>
<td>194.18</td>
<td>7.88%</td>
<td>1.03206</td>
<td>-29483</td>
<td>2.257E-03</td>
<td>-8.20%</td>
<td>2.1643E-04</td>
<td>3.65%</td>
</tr>
<tr>
<td>2</td>
<td>623.93</td>
<td>187.83</td>
<td>4.35%</td>
<td>1.00585</td>
<td>-2621</td>
<td>2.273E-03</td>
<td>0.93%</td>
<td>2.1384E-04</td>
<td>-1.20%</td>
</tr>
<tr>
<td>3</td>
<td>597.91</td>
<td>182.83</td>
<td>1.57%</td>
<td>1.00772</td>
<td>187</td>
<td>2.294E-03</td>
<td>0.96%</td>
<td>2.1432E-04</td>
<td>0.22%</td>
</tr>
<tr>
<td>4</td>
<td>588.67</td>
<td>180.46</td>
<td>0.26%</td>
<td>1.00991</td>
<td>219</td>
<td>2.304E-03</td>
<td>0.61%</td>
<td>2.1516E-04</td>
<td>0.39%</td>
</tr>
<tr>
<td>5</td>
<td>587.16</td>
<td>179.74</td>
<td>-0.14%</td>
<td>1.01122</td>
<td>131</td>
<td>2.307E-03</td>
<td>0.06%</td>
<td>2.1491E-04</td>
<td>-0.11%</td>
</tr>
<tr>
<td>6</td>
<td>588.01</td>
<td>179.62</td>
<td>-0.21%</td>
<td>1.01218</td>
<td>96</td>
<td>2.314E-03</td>
<td>0.03%</td>
<td>2.1403E-04</td>
<td>-0.41%</td>
</tr>
<tr>
<td>7</td>
<td>589.24</td>
<td>180.11</td>
<td>0.06%</td>
<td>1.01256</td>
<td>38</td>
<td>2.315E-03</td>
<td>-0.06%</td>
<td>2.1468E-04</td>
<td>0.31%</td>
</tr>
<tr>
<td>8</td>
<td>588.87</td>
<td>179.76</td>
<td>-0.13%</td>
<td>1.01178</td>
<td>-78</td>
<td>2.316E-03</td>
<td>-0.10%</td>
<td>2.1390E-04</td>
<td>-0.36%</td>
</tr>
<tr>
<td>9</td>
<td>589.66</td>
<td>180.40</td>
<td>0.22%</td>
<td>1.01168</td>
<td>-10</td>
<td>2.316E-03</td>
<td>0.01%</td>
<td>2.1432E-04</td>
<td>0.20%</td>
</tr>
</tbody>
</table>

Convergence after 8 or 9 iterations:
- $k_{\text{eff}}$ converged
- Nominal discharge burnup achieved
- Nuclide densities converged
2. Comparison of final core average fuel compositions

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Density [at/b-cm]</th>
<th>Relative difference</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Outer iteration 1</td>
<td>Outer iteration 2</td>
</tr>
<tr>
<td>xe-135</td>
<td>4.587E-08</td>
<td>4.422E-08</td>
</tr>
<tr>
<td>cs-134</td>
<td>1.542E-05</td>
<td>1.509E-05</td>
</tr>
<tr>
<td>cs-137</td>
<td>1.570E-04</td>
<td>1.568E-04</td>
</tr>
<tr>
<td>nd-148</td>
<td>4.405E-05</td>
<td>4.414E-05</td>
</tr>
<tr>
<td>sm-149</td>
<td>4.019E-07</td>
<td>4.122E-07</td>
</tr>
<tr>
<td>sm-151</td>
<td>1.856E-06</td>
<td>1.861E-06</td>
</tr>
<tr>
<td>gd-154</td>
<td>6.098E-08</td>
<td>6.104E-08</td>
</tr>
<tr>
<td>gd-155</td>
<td>2.865E-09</td>
<td>3.137E-09</td>
</tr>
<tr>
<td>eu-153</td>
<td>1.077E-05</td>
<td>1.065E-05</td>
</tr>
<tr>
<td>eu-154</td>
<td>1.788E-06</td>
<td>1.759E-06</td>
</tr>
<tr>
<td>eu-155</td>
<td>5.965E-07</td>
<td>5.876E-07</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th>Nuclide</th>
<th>Density [at/b-cm]</th>
<th>Relative difference</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Outer iteration 1</td>
<td>Outer iteration 2</td>
</tr>
<tr>
<td>u-235</td>
<td>2.316E-03</td>
<td>2.306E-03</td>
</tr>
<tr>
<td>u-238</td>
<td>1.786E-02</td>
<td>1.788E-02</td>
</tr>
<tr>
<td>pu-239</td>
<td>2.127E-04</td>
<td>2.143E-04</td>
</tr>
<tr>
<td>pu-240</td>
<td>8.041E-05</td>
<td>8.033E-05</td>
</tr>
<tr>
<td>pu-241</td>
<td>6.724E-05</td>
<td>6.662E-05</td>
</tr>
<tr>
<td>pu-242</td>
<td>2.980E-05</td>
<td>2.910E-05</td>
</tr>
<tr>
<td>am-241</td>
<td>6.746E-07</td>
<td>6.873E-07</td>
</tr>
<tr>
<td>cm-242</td>
<td>4.772E-07</td>
<td>4.672E-07</td>
</tr>
<tr>
<td>cm-244</td>
<td>1.467E-06</td>
<td>1.420E-06</td>
</tr>
</tbody>
</table>

Relative difference of core-average fuel composition is negligible besides very few exceptions in case of small nuclide densities.
3. Full core power profile

Results:
- Power is peaking in the inner fuel region.
- Consideration of axial/radial power profile in the iterations to obtain the equilibrium core compositions has minor effect.
4. Comparison of isothermal temperature coefficients

- Reactivity coefficient calculation:
  - $k_{\text{eff}}$ calculations with material temperatures varying over a range of several hundred K
  - Assuming constant temperature within material
  - Fitting of $\rho$ to determine coefficient

<table>
<thead>
<tr>
<th>Component</th>
<th>Temperature Reactivity Coefficient at HFP [pcm/K]</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Cisneros [1]</td>
</tr>
<tr>
<td>Fuel</td>
<td>-3.8</td>
</tr>
<tr>
<td>Salt coolant</td>
<td>-1.8</td>
</tr>
<tr>
<td>Graphite moderator</td>
<td>-0.7</td>
</tr>
<tr>
<td>Inner graphite reflector</td>
<td>+0.9</td>
</tr>
<tr>
<td>Outer graphite reflector</td>
<td>+0.9</td>
</tr>
</tbody>
</table>

6. Towards rapid inventory calculations with ORIGAMI

**Purpose of 1-group cross section analysis:** understand the spectral variations and their impact on 1-group cross sections which influence all inventory calculations

- Only small variation of 1-group removal cross section over depletion
- Small changes visible mainly in Pu-240

6. Comparison between UCB Mark 1 and PBMR-400

• Both cores showed significant radial variation for various nuclides
• Only UCB Mark 1 showed axial variation due to inlet/outlet geometry

7. Analytical model to calculate tritium production

- A simplified analytical model was developed by Cisneros et al* using a flux and one-group cross sections to allow estimation of tritium generation rates for an arbitrary initial Li-7 enrichment

\[
\dot{T}(t) = \phi \sigma_{Li-7}^T N_{Li-7} + \phi \sigma_{Li-6}^T \left( N_{Li-6}^0 e^{-\frac{V_{core}}{V_{loop}} \phi \sigma_{Li-6}^{abs} t} \right) + \frac{\phi \sigma_{Be-9}^\alpha N_{Be-9}}{\phi \sigma_{Li-6}^{abs}} \left( 1 - e^{-\frac{V_{core}}{V_{loop}} \phi \sigma_{Li-6}^{abs} t} \right)
\]

- SCALE results using TRITON/ORIGEN: 0.021 mol/day
- Equilibrium value from Cisneros analytical approach: 0.023 mol/day

---

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

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

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

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

Experimental Basis
- Oxidation/Gas Generation
- Melt Progression
- Fission Product Release
- Fission Product Transport

PIRT process

Accident Analysis
- Scenario # 1
- Scenario # 2
- Scenario # n-1
- Scenario # n

Synthesize timings and release fractions

Design Basis Source Term

MELCOR

Cesium release from high burnup fuel. Comparison to results of VERCORS Test

Cs Diffusivity

Synthesize

Fission Product classes
- Noble Gases
- Iodine, bromine
- Cesium
- Tellurium
- Ra, Sr
- Ru, Mo, Pd, etc.
- Lanthanides
- Cerium group

Scenario # n-1

Scenario # n
### SCALE
**Neutronics**
- Criticality
- Shielding
- Radionuclide inventory
- Burnup credit
- Decay heat

### MELCOR
**Integrated Severe Accident Progression**
- Hydrodynamics for range of working fluids
- Accident response of plant structures, systems and components
- Fission product transport

### MACCS
**Radiological Consequences**
- Near- and far-field atmospheric transport and deposition
- Assessment of health and economic impacts

---

**Nuclear Reactor System Applications**

<table>
<thead>
<tr>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td>Technology-neutral</td>
<td>License amendments</td>
<td>Design analysis scoping calculations</td>
<td>Neutron beam injectors</td>
<td>Risk studies</td>
<td>Risk studies</td>
<td>Leak path factor calculations</td>
</tr>
<tr>
<td>Experimental</td>
<td>Risk-informed regulation</td>
<td>Training simulators</td>
<td>Li loop LOFA transient analysis</td>
<td>Multi-unit accidents</td>
<td>Multi-unit accidents</td>
<td>DOE safety toolbox codes</td>
</tr>
<tr>
<td>Naval</td>
<td>Design certification (e.g., NuScale)</td>
<td></td>
<td>ITER cryostat modeling</td>
<td>Dry storage</td>
<td>Dry storage</td>
<td>DOE nuclear facilities</td>
</tr>
<tr>
<td>Advanced LWRs</td>
<td>Vulnerability studies</td>
<td></td>
<td>He-cooled pebble test blanket (H3)</td>
<td>Spent fuel transport/package applications</td>
<td>Spent fuel transport/package applications</td>
<td></td>
</tr>
<tr>
<td>Advanced Non-LWRs</td>
<td>Emergency preparedness</td>
<td></td>
<td></td>
<td></td>
<td></td>
<td>(Pantex, Hanford, Los Alamos, Savannah River Site)</td>
</tr>
<tr>
<td>Accident forensics</td>
<td>Emergency Planning Zone Analysis</td>
<td></td>
<td></td>
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<td></td>
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<tr>
<td>(Fukushima, TMI)</td>
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<tr>
<td>Probabilistic risk</td>
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<tr>
<td>assessment</td>
<td></td>
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<td></td>
<td></td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

---

**Non-Reactor Applications**

- Neutron beam injectors
- Li loop LOFA transient analysis
- ITER cryostat modeling
- He-cooled pebble test blanket (H3)
MELCOR Attributes
Foundations of MELCOR Development

Fully integrated, engineering-level code
• Thermal-hydraulic response of reactor coolant system, reactor cavity, reactor enclosures, and auxiliary buildings
• Core heat-up, degradation and relocation
• Core-concrete interaction
• Flammable gas production, transport and combustion
• Fission product release and transport behavior

Level of physics modeling consistent with
• State-of-knowledge
• Necessity to capture global plant response
• Reduced-order and correlation-based modeling often most valuable to link plant physical conditions to evolution of severe accident and fission product release/transport

Traditional application
• Models constructed by user from basic components (control volumes, flow paths and heat structures)
• Demonstrated adaptability to new reactor designs – HPR, HTGR, SMR, MSR, ATR, Naval Reactors, VVER, SFP,…
Validated physical models
- International Standard Problems, benchmarks, experiments, and reactor accidents
- Beyond design basis validation will always be limited by model uncertainty that arises when extrapolated to reactor-scale

Cooperative Severe Accident Research Program (CSARP) is an NRC-sponsored international, collaborative community supporting the validation of MELCOR

International LWR fleet relies on safety assessments performed with the MELCOR code
Common Phenomenology
Modeling is mechanistic consistent with level of knowledge of phenomena supported by experiments

Parametric models enable uncertainties to be characterized

• Majority of modeling parameters can be varied
• Properties of materials, correlation coefficients, numerical controls/tolerances, etc.

Code models are general and flexible

• Relatively easy to model novel designs
• All-purpose thermal hydraulic and aerosol transport code
MELCOR Software Quality Assurance – Best Practices

MELCOR SQA Standards
- SNL Corporate procedure IM100.3.5
- CMMI-4+
- NRC NUREG/BR-0167

MELCOR Wiki
- Archiving information
- Sharing resources (policies, conventions, information, progress) among the development team.

Code Configuration Management (CM)
- ‘Subversion’
- TortoiseSVN
- VisualSVN integrates with Visual Studio (IDE)

Reviews
- Code Reviews: Code Collaborator
- Internal SQA reviews

Continuous builds & testing
- DEF application used to launch multiple jobs and collect results
- Regression test report
- More thorough testing for code release
- Target bug fixes and new models for testing

Emphasis is on Automation
- Affordable solutions
- Consistent solutions

Bug tracking and reporting
- Bugzilla online

Code Validation
- Assessment calculations
- Code cross walks for complex phenomena where data does not exist.

Documentation
- Available on ‘Subversion’ repository with links from wiki
- Latest PDF with bookmarks automatically generated from word documents under Subversion control
- Links on MELCOR wiki

Project Management
- Jira for tracking progress/issues
- Can be viewable externally by stakeholders

Sharing of information with users
- External web page
- MELCOR workshops
- MELCOR User Groups (EMUG & AMUG)

Table 1-1: Physics Package Coverage
MELCOR Verification & Validation Basis

Volume 1: Primer & User Guide
Volume 3: MELCOR Assessment Problems
[SAND2015-6693 R]

Analytical Problems
Saturated Liquid Depressurization
Adiabatic Expansion of Hydrogen
Transient Heat Flow in a Semi-Infinite Heat Slab
Cooling of Heat Structures in a Fluid
Radial Heat Conduction in Annular Structures
Establishment of Flow

Specific to non-LWR applications

LWR & non-LWR applications

Sodium Fires (Completed)
Molten Salt (planned)
Sodium Reactors (planned)
HTGR (planned)

LOF, LOHS, TOP TREAT M-Series ANL-ART-38
Sample Validation Cases

TRISO Diffusion Release

IAEA CRP-6 Benchmark
Fractional Release

<table>
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<tr>
<th>Case</th>
<th>1a</th>
<th>1b</th>
<th>2a</th>
<th>2b</th>
<th>3a</th>
<th>3b</th>
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</table>

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

A sensitivity study to examine fission product release from a fuel particle starting with a bare kernel and ending with an irradiated TRISO particle;

LACE LA1 and LA3 tests experimentally examined the transport and retention of aerosols through pipes with high speed flow.

Aerosol Physics
- Agglomeration
- Deposition
- Condensation and Evaporation at surfaces

Validation Cases
- Simple geometry: AHMED, ABCOVE (AB5 & AB6), LACE(LA4),
- Multi-compartment geometry: VANAM (M3), DEMONA(B3)
- Deposition: STORM, LACE(LA1, LA3)

Resuspension

STORM (Simplified Test of Resuspension Mechanism) test facility

Cumulative Deposited Mass (g) vs. Distance From Pipe Entrance (m)
MELCOR Modernization

Generalized numerical solution engine

Hydrodynamics

In-vessel damage progression

Ex-vessel damage progression

Fission product release and transport

TP = Transfer Process
DCH = Decay Heat
COR = Core
SPR = Confinement Spray
BUR = Gas Bubble
FDI = Fuel Dispersed Interaction
CAV = Cavity (MCC)
ESF = Engineering Safety Features
MP = Material Properties
RN = Reactor
MES = Material Properties
EDF = Emergency Data File
CF = Control Function
MV = Special Messages
NCG = Non Condensable Gas

Separate Physics & Numerics

Material Properties & EOS
MELCOR provides mass of radionuclides released into salt, chemistry, T and P

Chemistry model computes what remains in salt as soluble, colloidal, deposited, and released as vapor and aerosol

MELCOR continues to transport materials to and from the salt control volume

Each Timestep
Cs vapor pressures in MSM calculations