Nuclear data assessment for non-LWRs with SCALE

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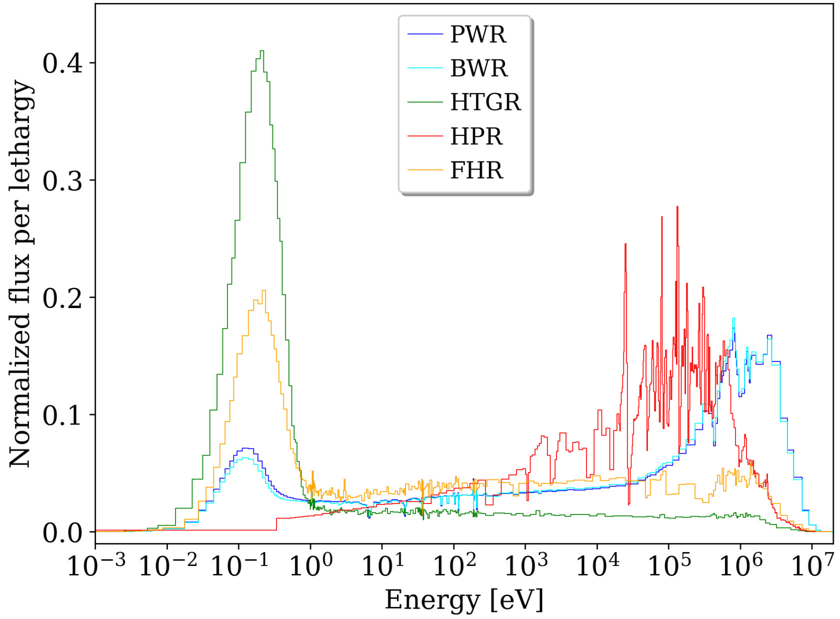
Nuclear data performance assessments for non-LWR

- Project goal: Identification of key nuclear data and nuclear data uncertainties for the analysis of key metrics for reactor safety of non-LWRs

- Relevant non-LWRs designs:
  - Heat pipe reactor (HPR): INL Design A
  - High temperature gas-cooled reactor (HTGR): HTR-10
  - Molten salt reactor (MSR): MSRE
  - Fluoride salt-cooled HTR (FHR): UC Berkeley PB-FHR Mark 1
  - Sodium-cooled fast reactor (SFR): MET1000 and EBR-II



- Approach:
  - Identify data gaps in nominal and uncertainty data based on literature review and previous ORNL studies
  - Interrogate ENDF/B-VII.0, VII.1 and VIII.0 libraries to identify relevant updates in nominal values and uncertainties
  - Propagate uncertainties to key figures of merit relevant of reactor safety

- Results: F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, W. A. Wieselquist (2021), "Nuclear Data Assessment for Advanced Reactors", NUREG/CR-7289, https://www.nrc.gov/docs/ML2134/ML21349A369.pdf

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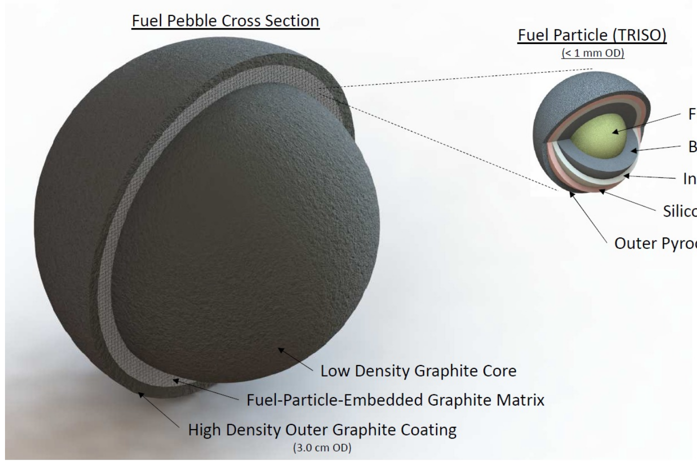
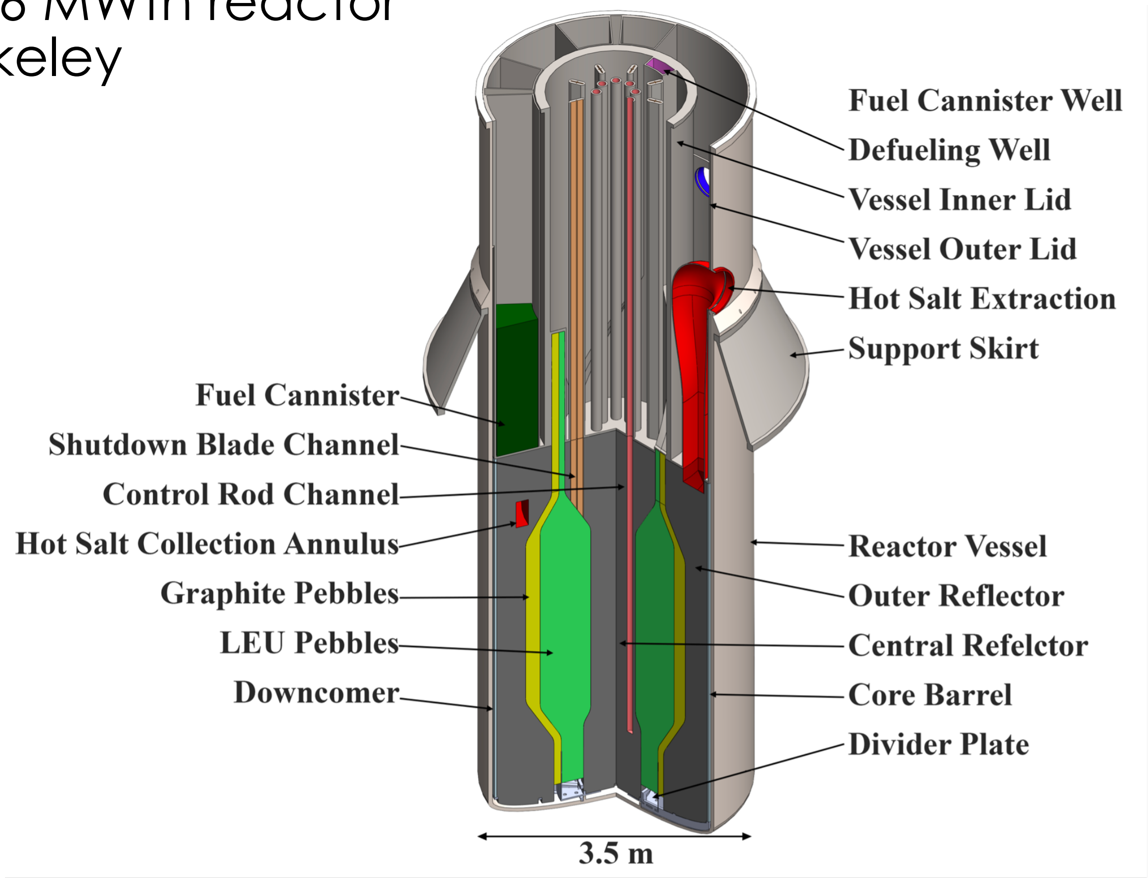
Example 1: FHR – UC Berkeley Pebble-bed FHR

- Pebble-bed fluoride salt-cooled high temperature reactor
- Preconceptual design for a small, modular 236 MWth reactor developed by the University of California, Berkeley
- Relevant characteristics:
  - Fuel: TRISO particles with UCO fuel kernels
  - Coolant: FLiBe salt
  - Moderator: graphite



TRISO particle and FHR fuel pebble

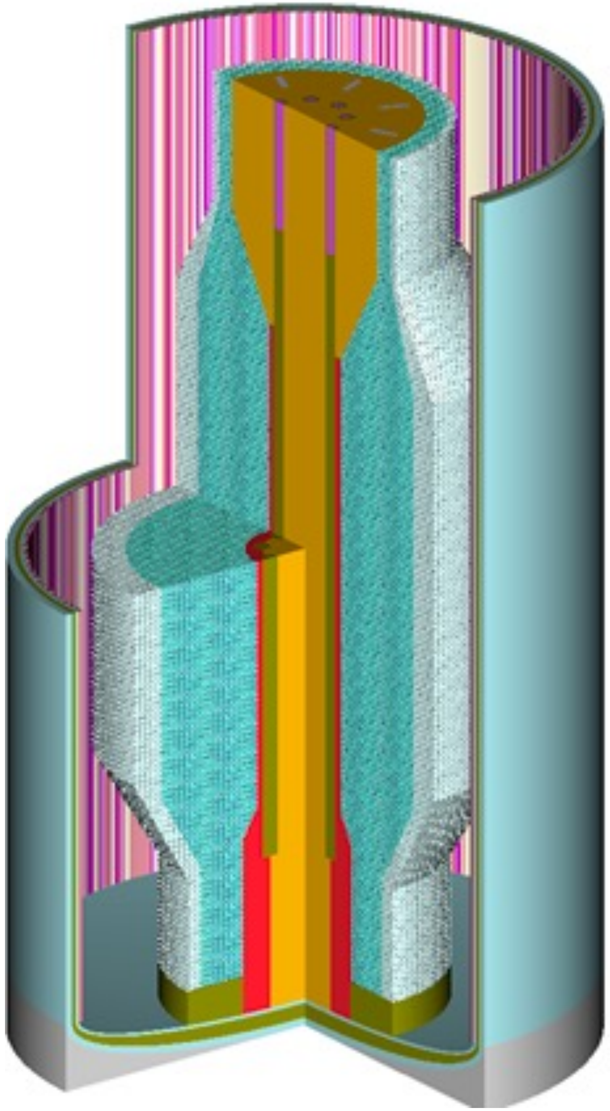


UC Berkeley PB-FHR

Ref.: C. Andreades et al., "Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant," Berkeley, CA, UCBTH-14-002, 2014.

FHR: Modeling with SCALE

- Neutron transport: KENO-VI Monte Carlo
- Uncertainty analysis: Sampler (sample size: 1,000)
- Cross section library: 252-group
- Covariance library: 56-group
- Quantities of interest:
  - Multiplication factor k_{eff}
  - Fuel temperature reactivity (ΔT=500K)
  - Salt temperature reactivity (ΔT=300K)
  - Salt density reactivity (density x1.5)
  - Axial power distribution

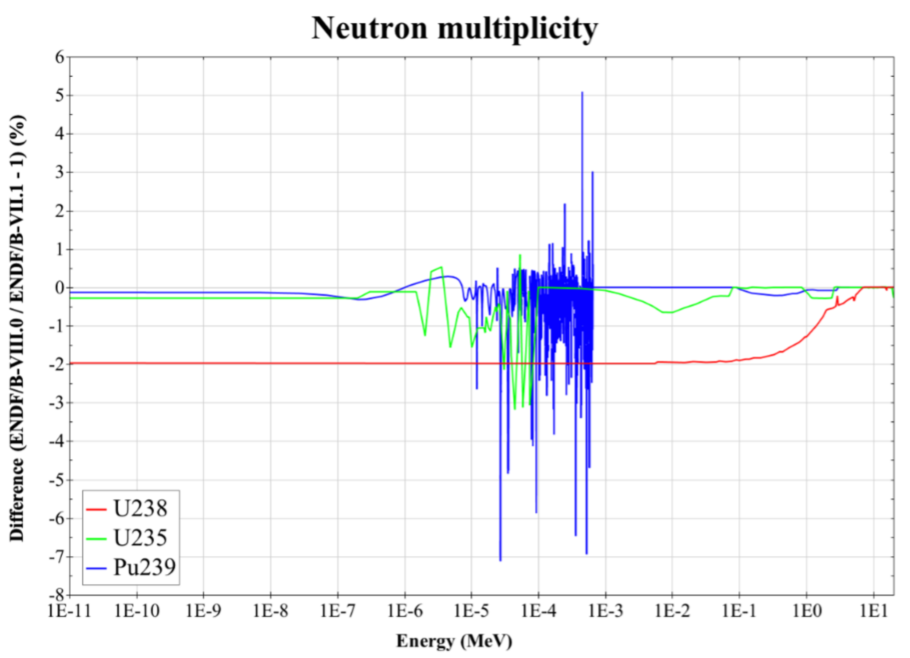


SCALE model of the UCB FHR

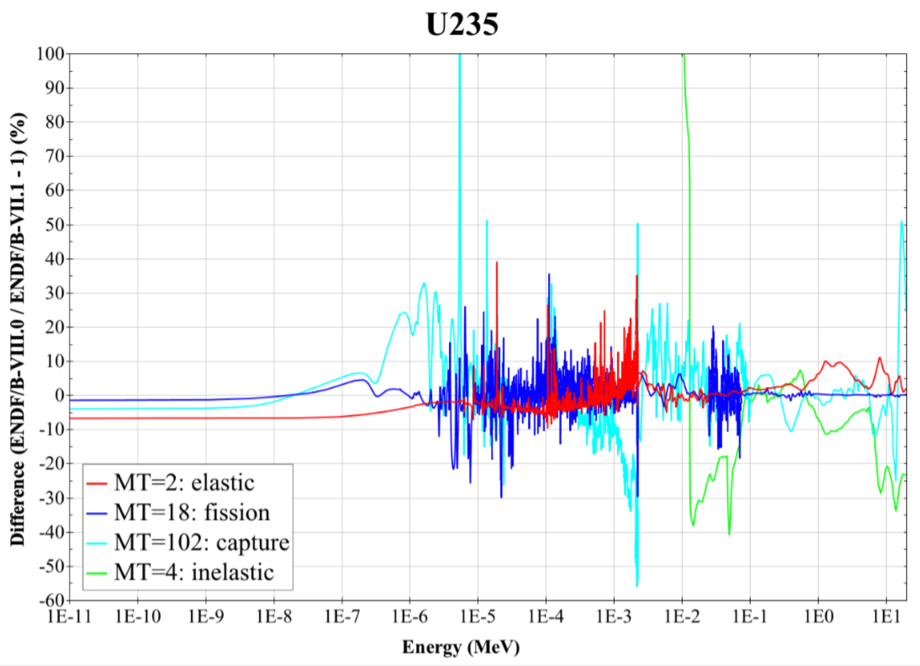
FHR: Effect of cross sections on nominal reactivity values

Quantity	ENDF/B-VII.1	ENDF/B-VIII.0	Δρ [pcm]
k_{eff}	0.94091 ± 0.00008	0.94368 ± 0.00021	312 ± 25
Fuel temp. reactivity	-776 ± 12	-745 ± 34	30 ± 36
Salt temp. reactivity	-363 ± 11	-373 ± 36	-10 ± 38
Salt density reactivity	-1071 ± 12	-1015 ± 44	56 ± 45

Reactivity difference between libraries are due to changes in relevant cross sections including ^{235}U and ^{238}U.



Difference: ENDF/B-VIII.0 vs. ENDF/B-VII.1

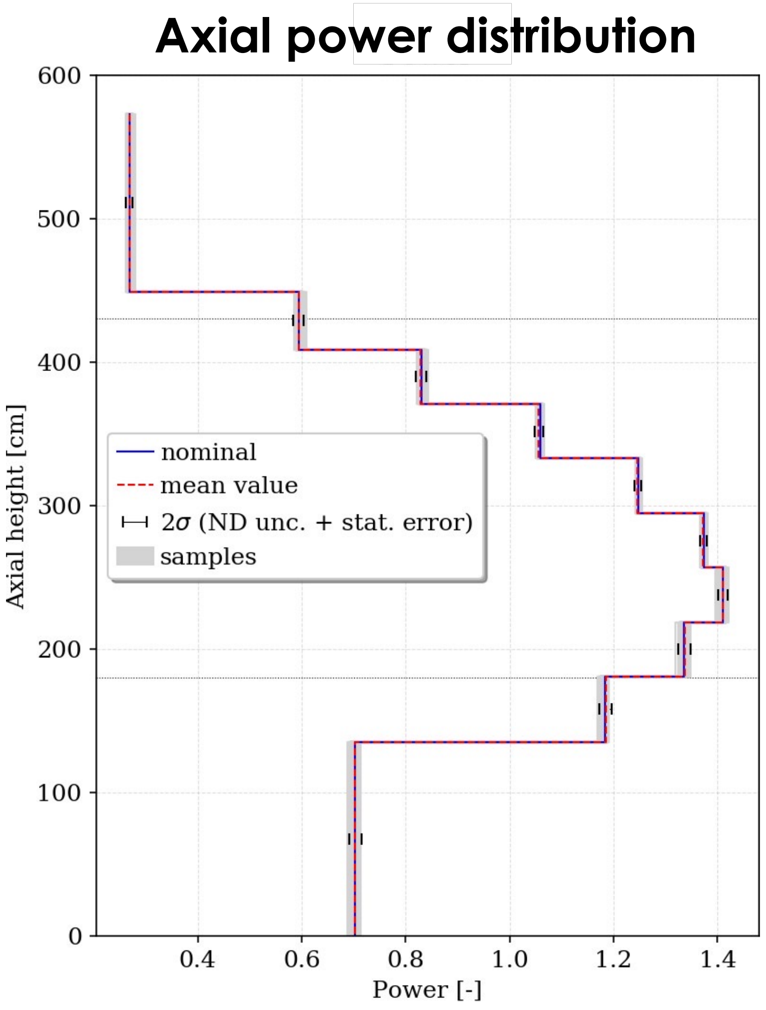


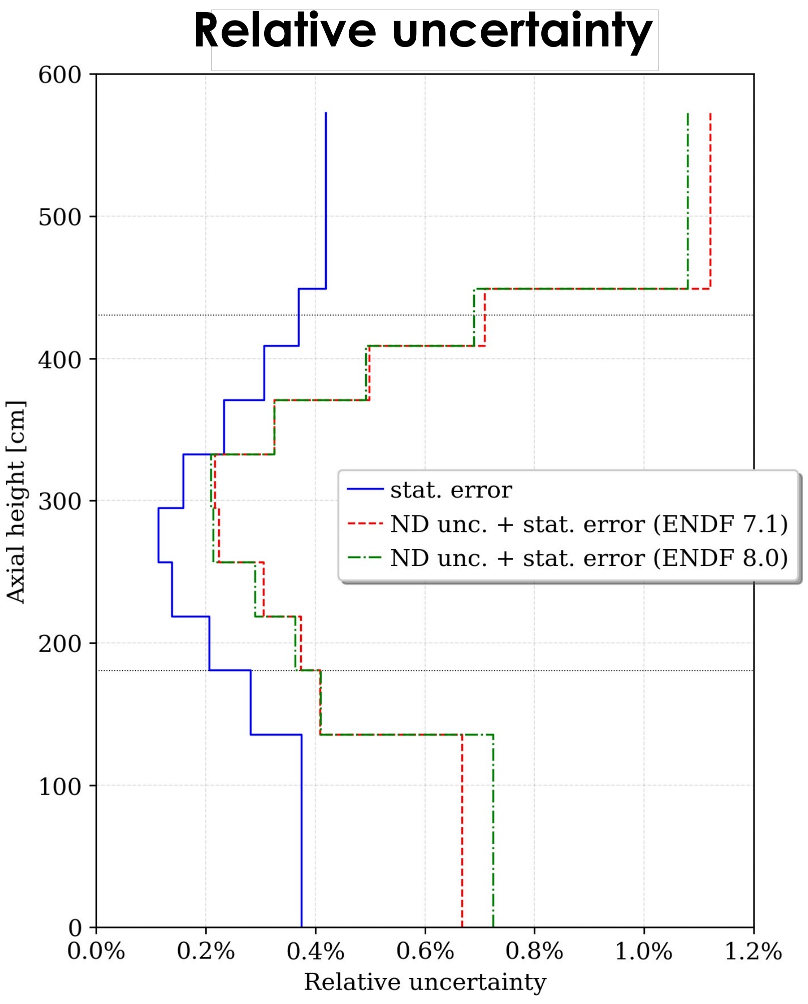
FHR: Uncertainties in reactivity due to nuclear data uncertainties

Quantity	ENDF/B-VII.1	ENDF/B-VIII.0	Rel. difference
k_{eff}	1.38%	1.43%	3.60%
Fuel temperature reactivity	3.11%	2.79%	-10.17%
Salt temperature reactivity	5.54%	7.13%	28.68%
Salt density reactivity	35.65%	36.80%	3.22%

- Uncertainties due to nuclear data vary significantly between libraries.
- Both absolute and relative uncertainty must be considered:
  - 1.4% uncertainty in k_{eff} corresponds to ~1,300 pcm
  - 36% uncertainty in the salt density corresponds to ~130 pcm

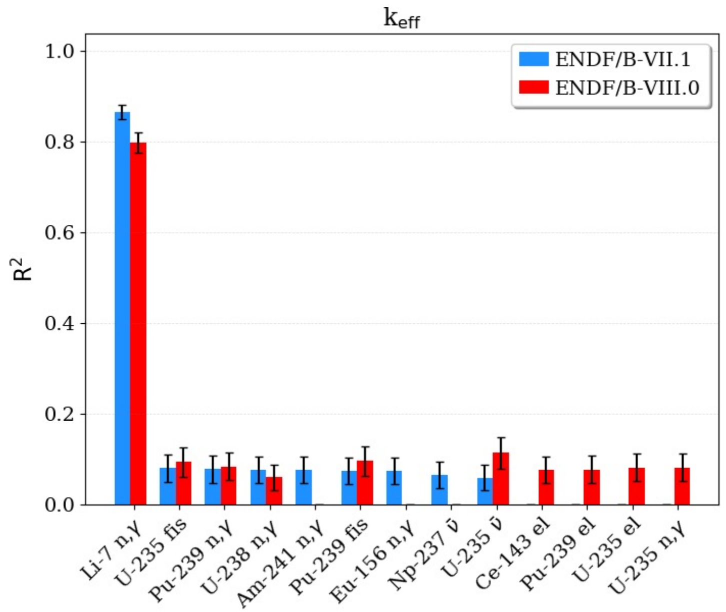
FHR: Power distribution and uncertainty

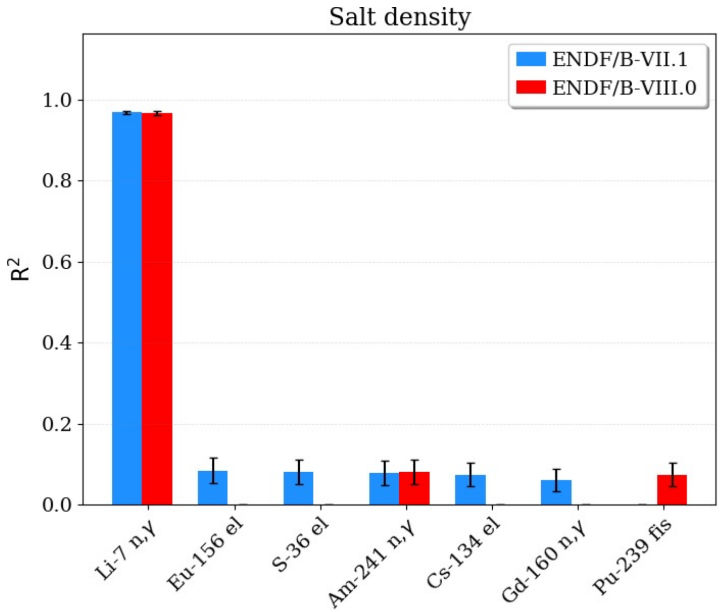


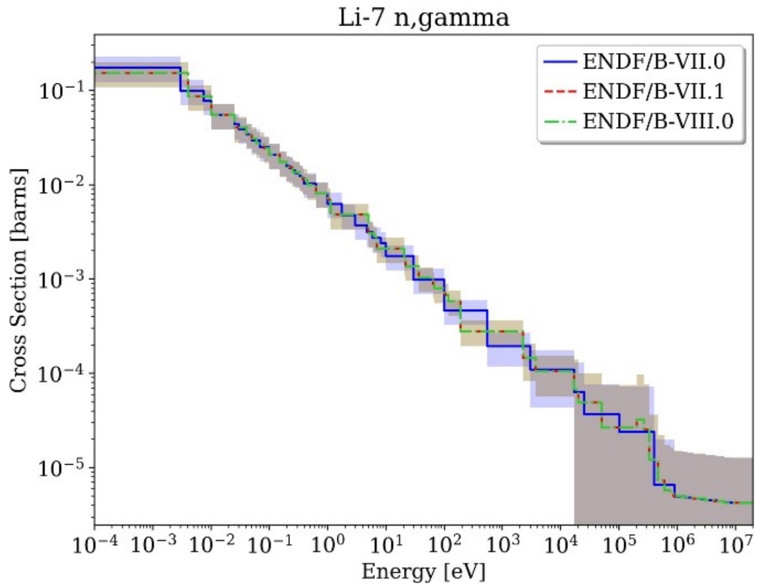


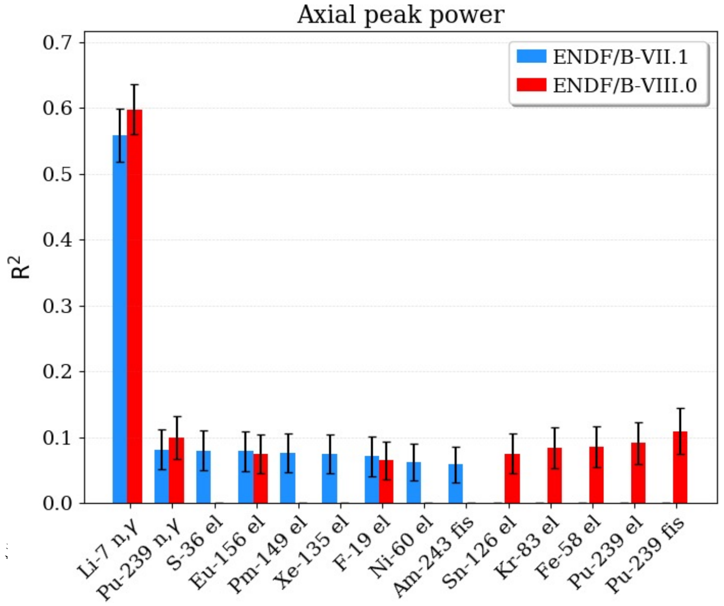
- Statistical uncertainty of axial power determined by repeating Monte Carlo calculations with different random seeds
- Peak power statistical error: ~0.14%
- Peak power uncertainty: 0.305% and 0.280%

FHR: Top contributors to uncertainty

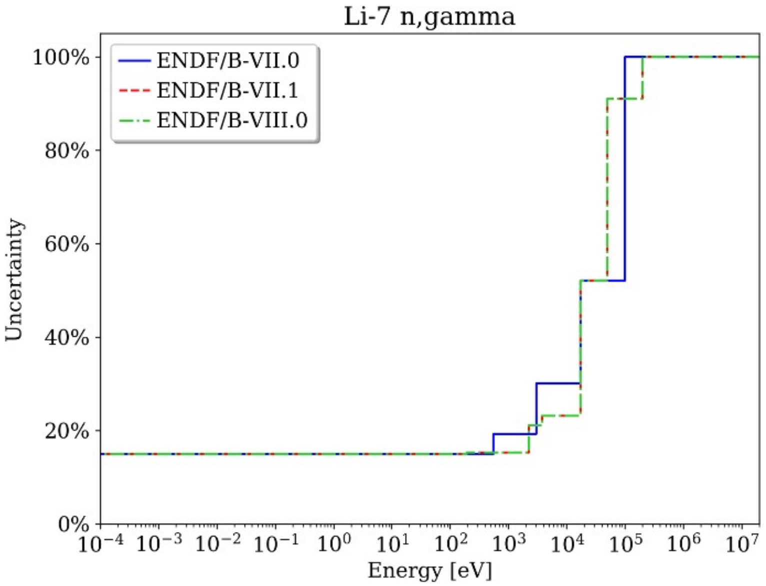






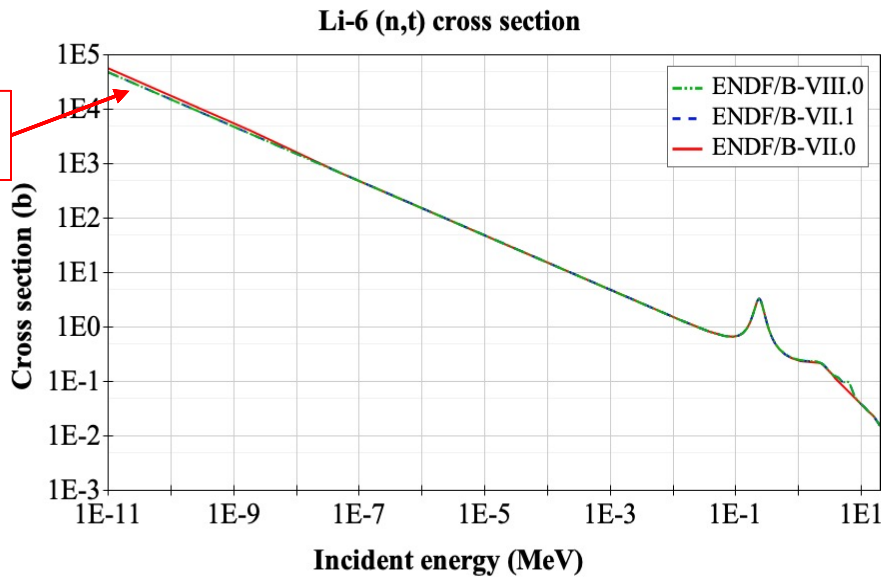


- Largest contributor to uncertainty of most analyzed quantities is **^{7}Li (n,γ)**
- The uncertainty in many quantities obtained for the FHR can be largely reduced if this uncertainty is reduced.

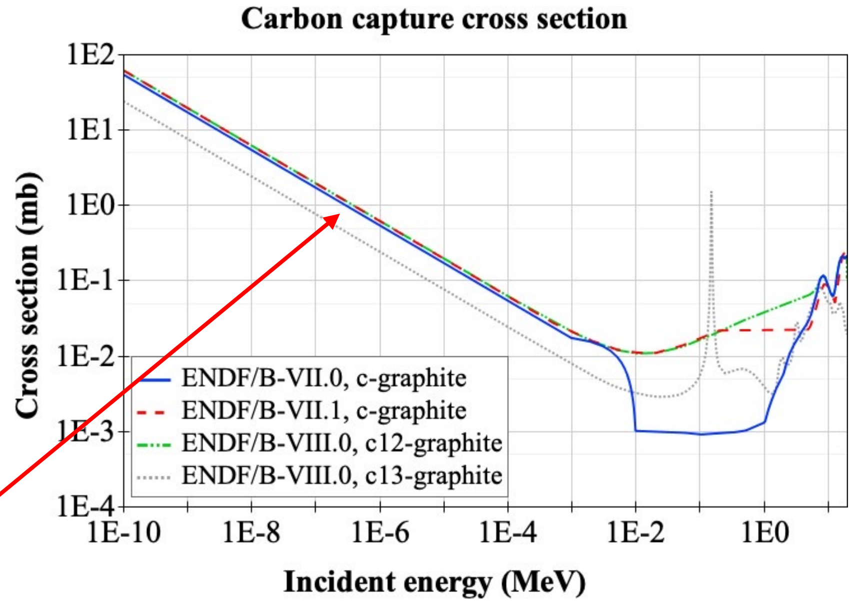


Nuclear data assessment for FHR

~50% difference between VII.0 and VII.1



- Key nominal data:
  - $\bar{\nu}$, fission, and the (n,γ) reaction of the major fissile isotopes ^{235}U, ^{239}Pu, and ^{241}Pu.
  - ^{6}Li (n,t): significant cross section update from ENDF/B-VII.0 to ENDF/B-VII.1
  - Carbon (n,γ): significant cross section update from ENDF/B-VII.0 to ENDF/B-VII.1
- Nuclear data uncertainties:
  - Large uncertainty of ^{7}Li (n,γ) is the dominating contributor to most reactivity effects considered here
  - Gap identified in availability of uncertainty data for thermal scattering data for both graphite and FLiBe



~15% difference between VII.0 and VII.1

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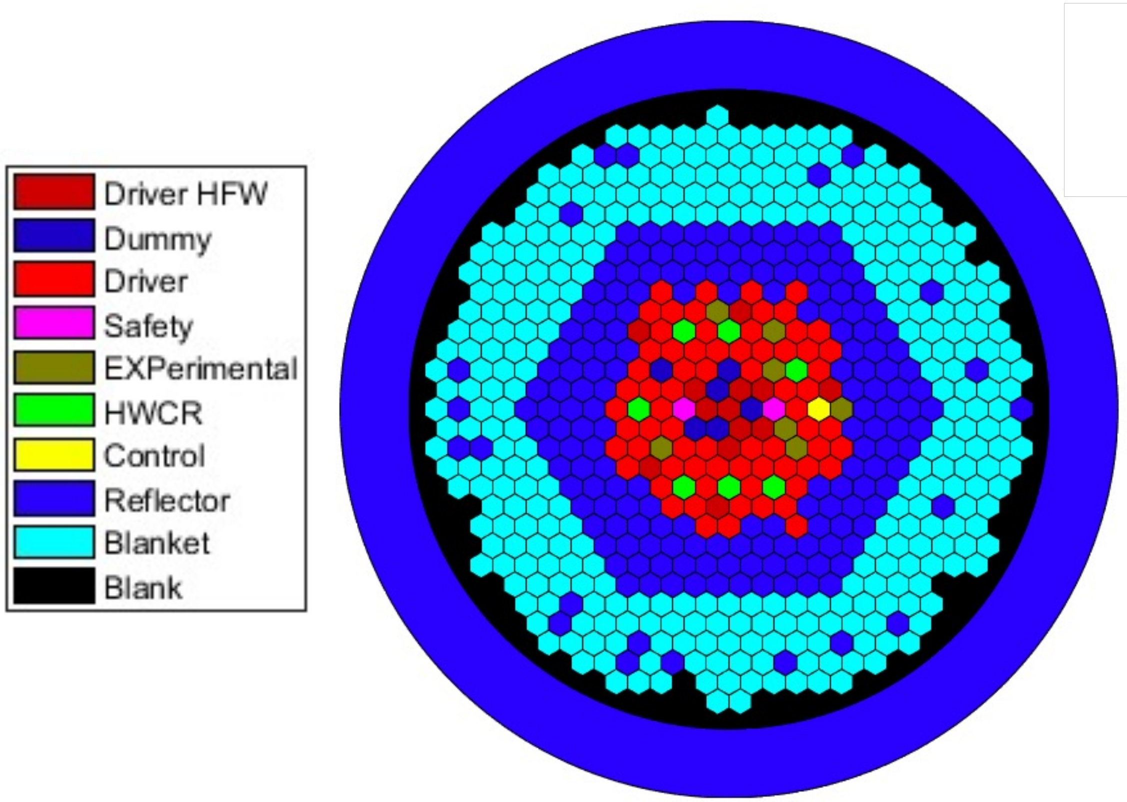
Example 2: SFR – Experimental Breeder Reactor II

- Operated by Argonne National Laboratory (now on the site of Idaho National Laboratory) between 1964 and 1994
- Evaluation of EBR-II run 138B, a test within the Shutdown Heat Removal Tests series conducted on 3 April 1986, was recently included in the International Handbook of Evaluated Reactor Physics Benchmark Experiments
- Measured data is limited to k_{eff}
- Relevant characteristics:
  - Fuel: high enriched uranium fuel and depleted uranium blanket assemblies
  - Coolant: sodium
  - Structure: steel

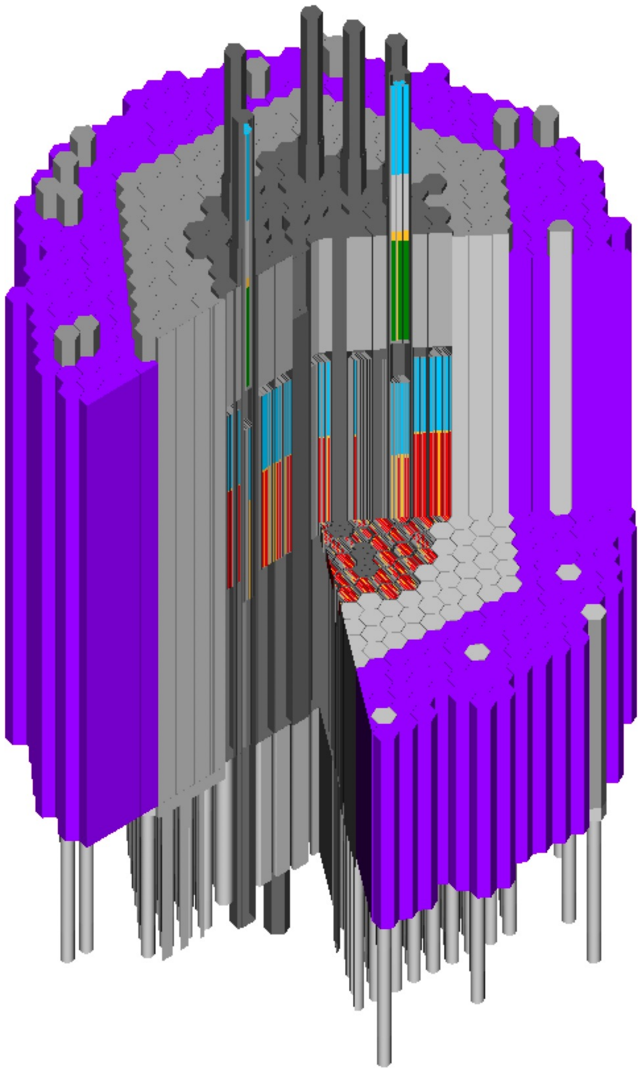


EBR-II core

E. S. Lum, et al., Evaluation of Run 138B at Experimental Breeder Reactor II, a Prototypic Liquid Metal Reactor; EBR2-LMFR-RESR-001, CRIT, NEA/NSC/DOC(2006)1; OECD/NEA, 2018.

SFR: Modeling with SCALE

- Neutron transport: KENO-VI Monte Carlo
- Uncertainty analysis: TSUNAMI and Sampler (sample size: 500)
- Cross section library: continuous-energy and 302-group
- Covariance library: 56-group
- Quantities of interest:
  - Multiplication factor k_{eff}
  - Control rod worth
  - Sodium void worth



SCALE model of the EBR-II

SFR: Nominal reactivity values

Comparison calculation to measurement

	k_{eff}	Δk [pcm]
Benchmark*	1.00927 ± 0.00618	(ref)
ENDF/B-VII.1	1.00703 ± 0.00016	-224 ± 618
ENDF/B-VIII.0	1.00704 ± 0.00019	-223 ± 618

Impact of nuclear data evaluation

	ENDF/B-VII.1	ENDF/B-VIII.0	$\Delta\rho$ [pcm]
k_{eff}	1.00703 ± 0.00016	1.00704 ± 0.00019	1 ± 25
CR worth [pcm]	4728 ± 26	4728 ± 25	0 ± 36
Na void [pcm]	-4651 ± 18	-4681 ± 29	-29 ± 34

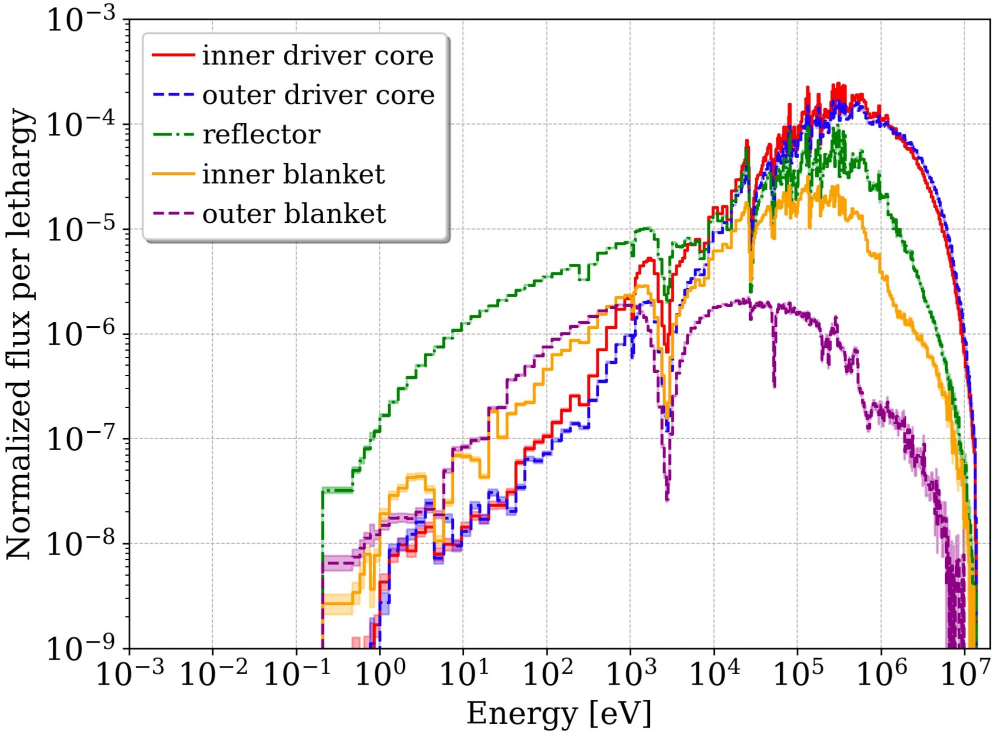
- All calculated results agree with measured k_{eff} considering benchmark uncertainty.
- Differences between ENDF/B-VII.1 and ENDF/B-VIII.0 results negligible.

* E. S. Lum, et al., Evaluation of Run 138B at Experimental Breeder Reactor II, a Prototypic Liquid Metal Reactor; EBR2-LMFR-RESR-001, CRIT, NEA/NSC/DOC(2006)1; OECD/NEA, 2018.

SFR: Uncertainties in reactivity due to nuclear data uncertainties

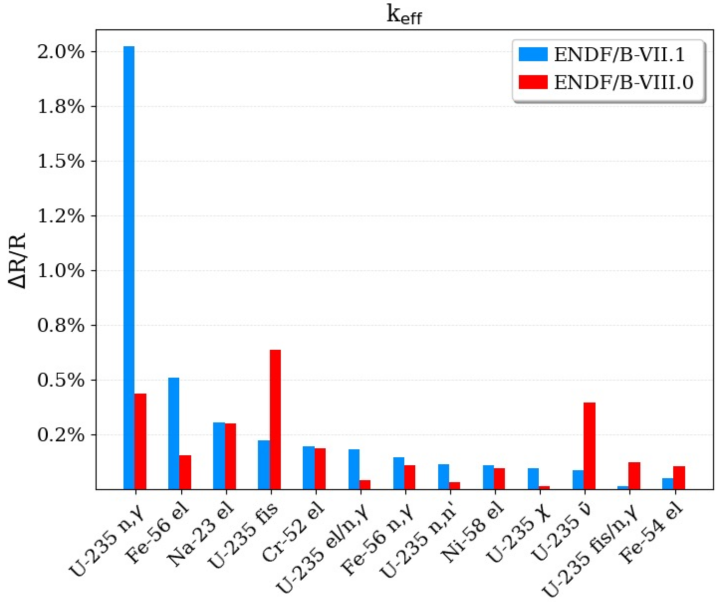
	ENDF/B-VII.1	ENDF/B-VIII.0	Rel. difference
k_{eff}	2.16%	1.01%	-53.0%
CR worth	1.18%	1.26%	6.8%
Na void	5.15%	4.21%	-18.3%

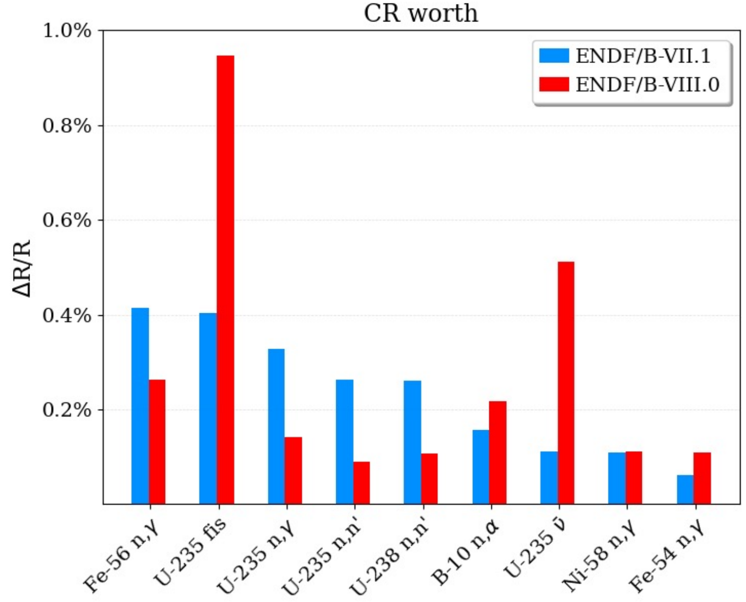
Significantly reduced k_{eff} uncertainties in ENDF/B-VIII.0.

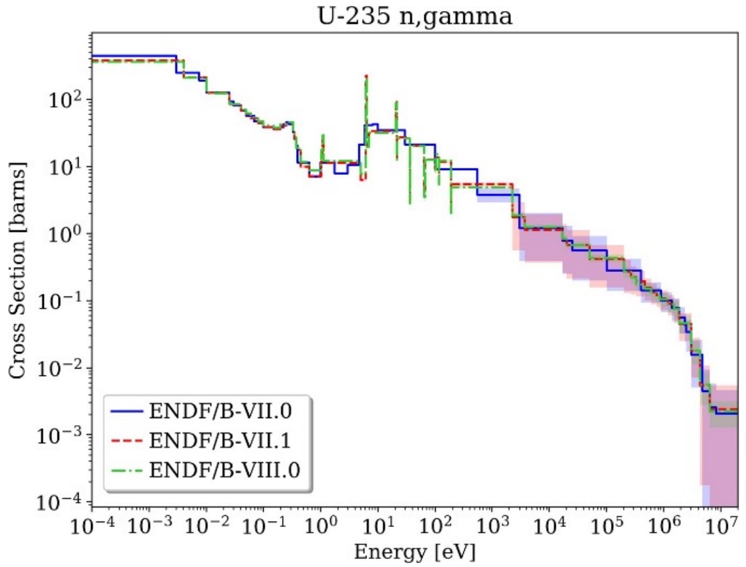


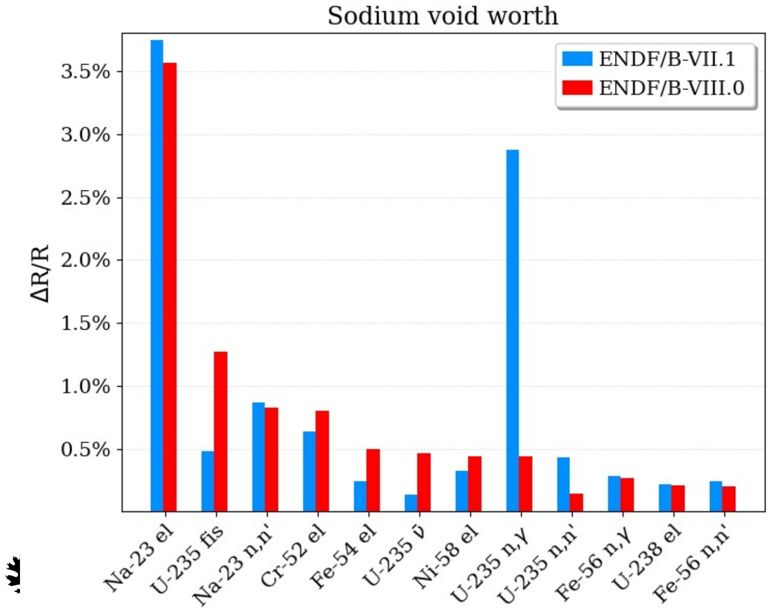
Neutron flux spectra in different regions

SFR: Top contributors to reactivity uncertainties

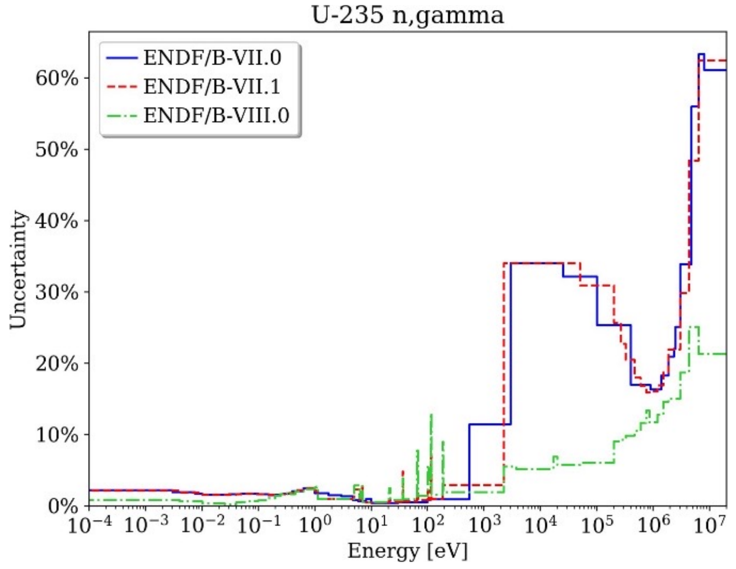






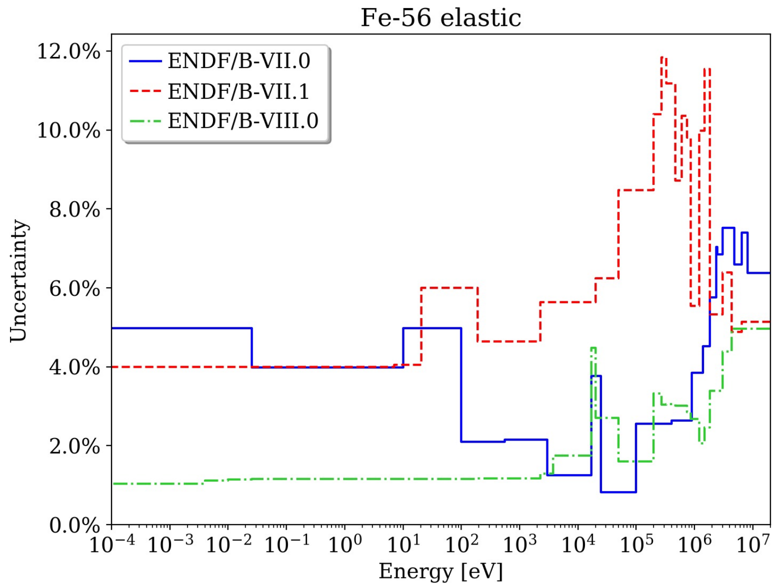


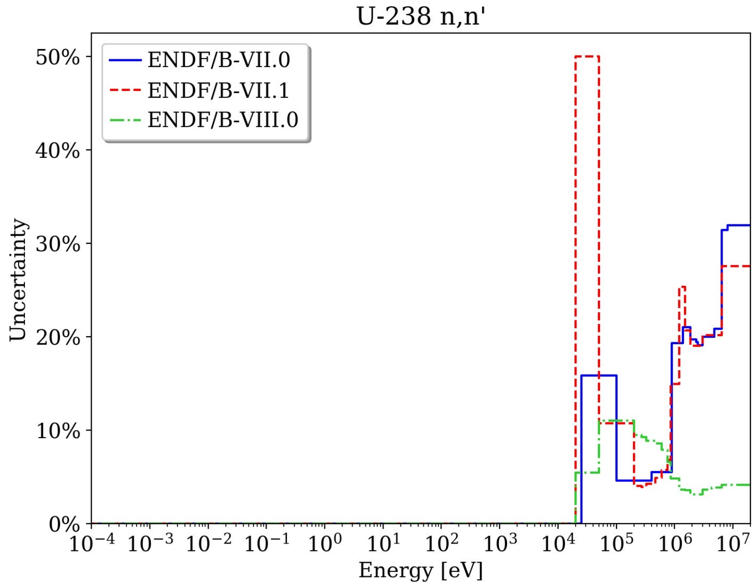
- Major influence of uncertainty in ^{235}U (n,γ)
- Reduced uncertainty of this reaction in ENDF/B-VIII.0 compared to ENDF/B-VII.1 cuts k_{eff} uncertainty in half



Nuclear data assessment for high ^{235}U enriched SFR

- Key nominal data:
  - Fission, (n,γ), scattering, and $\bar{\nu}$, of ^{235}U, ^{238}U, ^{239}Pu, ^{240}Pu, and ^{241}Pu
  - The scattering and (n,γ) reactions of ^{56}Fe as part of the structural material and ^{23}Na as the coolant
- Nuclear data uncertainties:
  - Large uncertainty of ^{235}U (n,γ) is the dominating contributor to most reactivity effects considered here
  - Note: for SFRs containing mixed U/Pu fuel, the ^{238}U inelastic scattering update from ENDF/B-VII.1 to ENDF/B-VIII.0 plays a major role.
  - Gap identified in availability of uncertainty data for angular scattering distributions





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Key observations for all studied non-LWRs

- **All non-LWRs:**
  - Large differences between ENDF/B library releases for relevant nominal and uncertainty data: neutron multiplicity, fission, capture, scattering for ^{235}U, ^{238}U, major Pu isotopes
- **FHR:**
  - No graphite thermal scattering data uncertainties
  - No thermal scattering data for salts
  - Carbon (n,γ): significant update from ENDF/B-VII.0 to VII.1
  - Large ^{7}Li (n,γ) uncertainty
  - ^{6}Li (n,t): significant cross section update from ENDF/B-VII.0 to VII.1
- **HPR and SFR:**
  - No angular scattering uncertainties
  - Large ^{235}U (n,γ) uncertainty for HEU fuel
  - Large ^{238}U inelastic scattering uncertainty in case of U/TRU fuel
  - Large impact of scattering reactions of coolant and structural materials
- **HTGR:**
  - Carbon (n,γ): significant update from ENDF/B-VII.0 to VII.1
  - No graphite thermal scattering data uncertainties
- **Graphite-moderated MSR:**
  - No data for ^{135m}Xe
  - No thermal scattering data for salts
  - No graphite thermal scattering data uncertainties
  - Large ^{7}Li (n,γ) uncertainty
  - ^{6}Li (n,t): significant cross section update from ENDF/B-VII.0 to VII.1
- **Fast spectrum MSR:**
  - ^{35}Cl (n,p): significant cross section update from ENDF/B-VII.0 to VII.1
  - Large impact of ^{24}Mg elastic scattering uncertainty

Also many lessons were learnt for the application of our UQ/SA tools in full core analyses.

SCALE tools for nuclear data sensitivity and uncertainty analysis – Updates in SCALE 6.3

- TSUNAMI-3D enabled with the new Monte Carlo code Shift:
  - IFP method
  - Implemented to work in parallel (large memory footprint divided between processors)
- Sampler includes a new **`analysis`** block that enables the calculation of:
  - Pearson correlation coefficients and covariances between responses
  - Correlation and covariance matrices between responses
  - Sensitivity index R^2 for a list of provided nuclides
- ORIGEN sensitivity capability:
  - New **`sens`** block within the ORIGEN input
  - Determine the change in a particular nuclide or group of nuclides to all nuclear data involved in the calculation
  - Automatic direct perturbation cases to verify sensitivities

References and acknowledgements

- Support for this work was provided by the US NRC

- References for models and results: https://www.ornl.gov/scale/references
- Public repository of input files: https://code.ornl.gov/scale/analysis/non-lwr-models-vol3
- Major publications based on the presented project:

  – F. Bostelmann, G. Ilas, C. Celik, A. M. Holcomb, W. A. Wieselquist (2021), "Nuclear Data Assessment for Advanced Reactors", NUREG/CR-7289, https://www.nrc.gov/docs/ML2134/ML21349A369.pdf

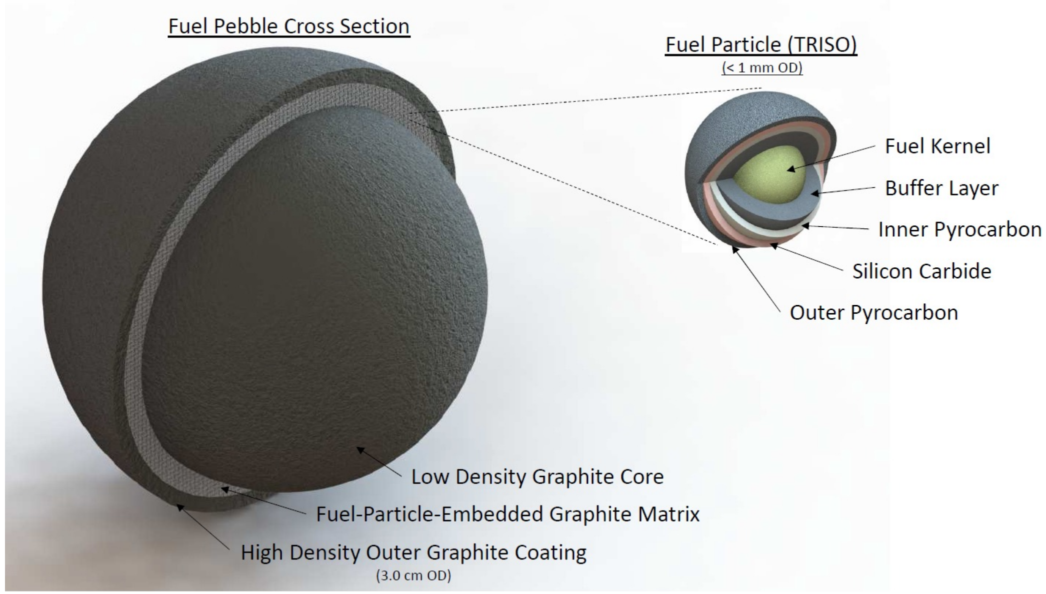
  – F. Bostelmann, S. E. Skutnik, E. Walker, G. Ilas, and W. A. Wieselquist (2021), "Modeling of the Molten Salt Reactor Experiment with SCALE," Nucl. Technol. doi:10.1080/00295450.2021.1943122

  – F. Bostelmann, G. Ilas, and W. A. Wieselquist (2021), "Nuclear Data Sensitivity Study for the EBR-II Fast Reactor Benchmark Using SCALE with ENDF/B-VII.1 and ENDF/B-VIII.0," J. Nucl. Eng., 2(4), 345–367. doi:10.3390/jne2040028

Backup

Example 1: FHR – UC Berkeley Pebble-bed FHR

- Pebble-bed fluoride salt-cooled high temperature reactor
- Preconceptual design for a small, modular 236 MWth reactor developed by the University of California, Berkeley
- No operating experience with this type of FHR reactor
- Relevant characteristics:
  - Fuel: TRISO particles with UCO fuel kernels
  - Coolant: FLiBe salt
  - Moderator: graphite

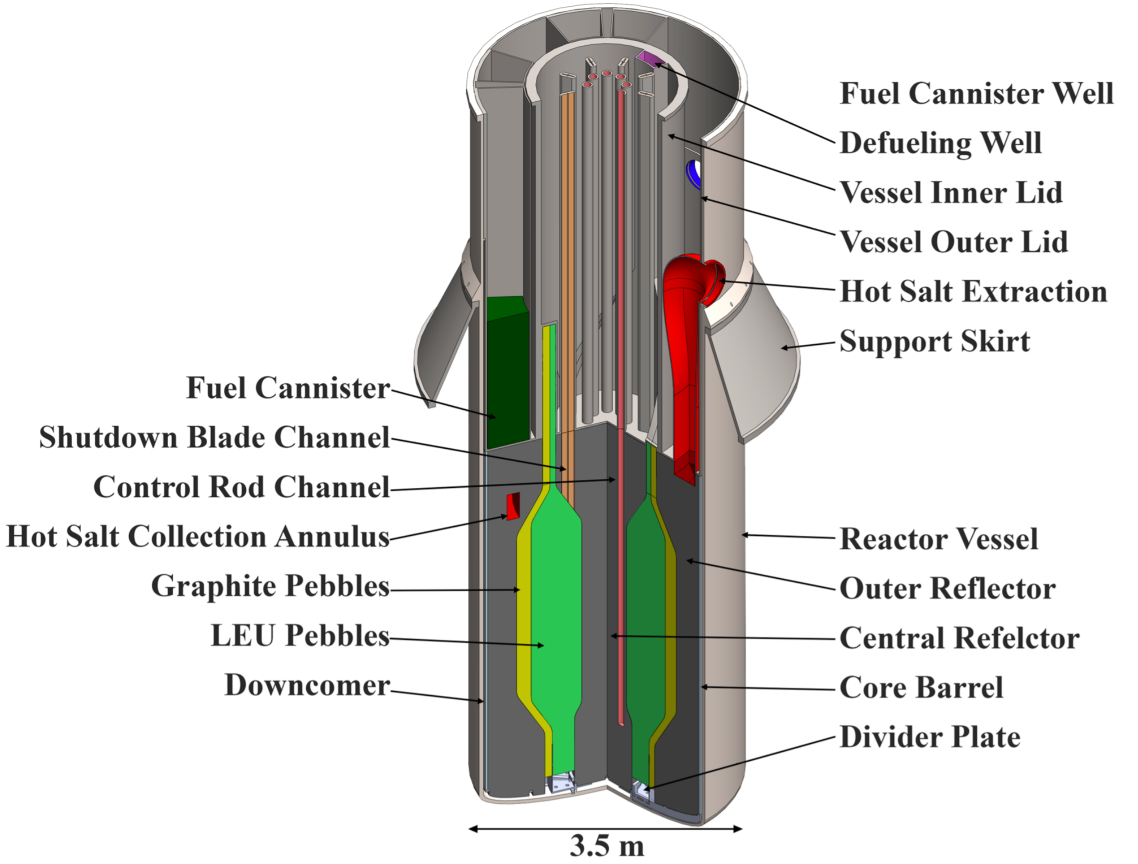


TRISO particle and FHR fuel pebble

Technical details: C. Andreades et al., "Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant," Berkeley, CA, UCBTH-14-002, 2014.
Figure: http://fhr.nuc.berkeley.edu/wp-content/uploads/2014/10/PEBBLE-SCHEMATIC-V2.png

Example 1: FHR – UC Berkeley Pebble-bed FHR

Description	Value
Reactor power	236 MWth
UCO fuel density	10.5 g/cc
Uranium enrichment	**19.9 wt.%**
Fuel kernel radius	0.2 mm
Number of particles in pebble	4,730
Particle packing fraction in fuel pebble	**40%**
Radius of fuel pebble	1.5 cm
Inner/outer radius of fuel zone	1.25/1.40 cm
Number of fuel pebbles	470,000
Number of unfueled/graphite pebbles	218,000
Pebble packing fraction	**60%**
Core Inner reflector radius	35 cm
Outer fuel pebble region radius	105 cm
Outer graphite pebble region radius	125 cm
Volume of active fuel region	10.4 m^3
Average pebble thermal power	500 W
Average pebble discharge burnup	**180 GWd/MTIHM**
Average pebble full-power lifetime	1.40 years



UC Berkeley PB-FHR

C. Andreades et al., "Technical Description of the "Mark 1" Pebble-Bed Fluoride-Salt-Cooled High-Temperature Reactor (PB-FHR) Power Plant," Berkeley, CA, UCBTH-14-002, 2014.

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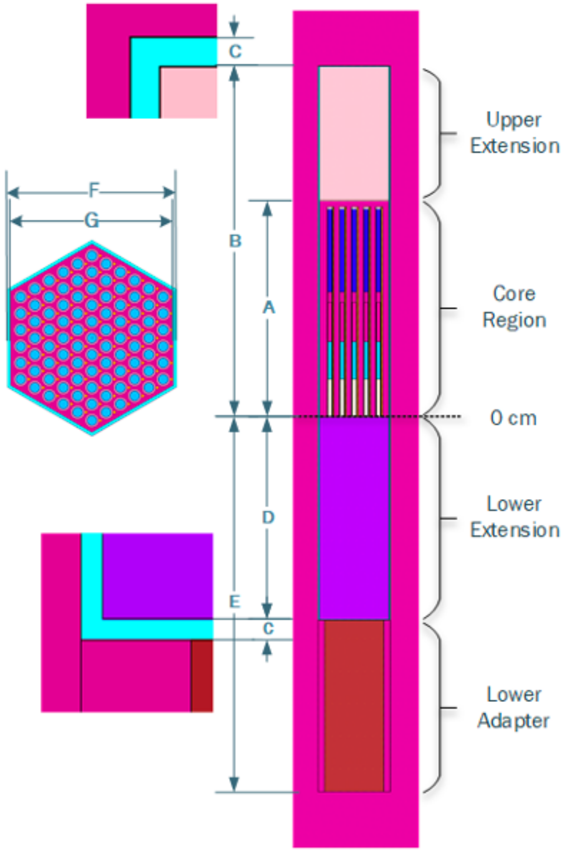
SFR: Main characteristics

Description	**Value**
Reactor power	62 MWth
Fuel material	**high enriched uranium metal**
Coolant material	**sodium**
Major structural material	**steel**
Temperature of all materials (K)	616
Number of fuel assemblies in the core:	
Full worth	70
Half worth	13
Number of fuel pins per assembly	91
Number of depleted uranium blanket assemblies	330
Assembly pitch (cm)	6.8877
Outer fuel radius (cm)	0.1651
Outer cladding radius (cm)	0.2210
Inner cladding radius (cm)	0.1905
Fuel pin pitch (cm)	0.566
Active core height (cm)	34.6075



EBR-II fuel assembly

E. S. Lum, et al., Evaluation of Run 138B at Experimental Breeder Reactor II, a Prototypic Liquid Metal Reactor; EBR2-LMFR-RESR-001, CRIT, NEA/NSC/DOC(2006)1; OECD/NEA, 2018.