ADVANCED REACTOR SAFEGUARDS & SECURITY

Uncertainty Quantification of Pebble's Discharge Burnup and Isotopic Inventory: Correlation Matrix

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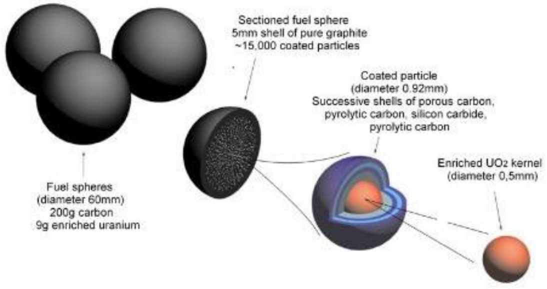
Introduction and Objective

- In a PBR, the **fuel burnup** of pebbles and **fissile content** in them depend on the several operating parameters and their variabilities.
  - neutron flux magnitude and neutron spectra
  - Initial ^{235}U enrichment, temperature, residence time
  - multi-pass scheme and the “channels” through the pebble passes
- Initial estimations of fissile content in the spent pebble by both the neutronics code and fuel burnup measurement are expected to be reasonable, however, will have opportunities for improvement.
  - Understanding the parameters/assumptions, most sensitive to the variability in fuel burnup and fissile content values, will inform reactor modelers and designers as to where to target improvements to get more accurate values.



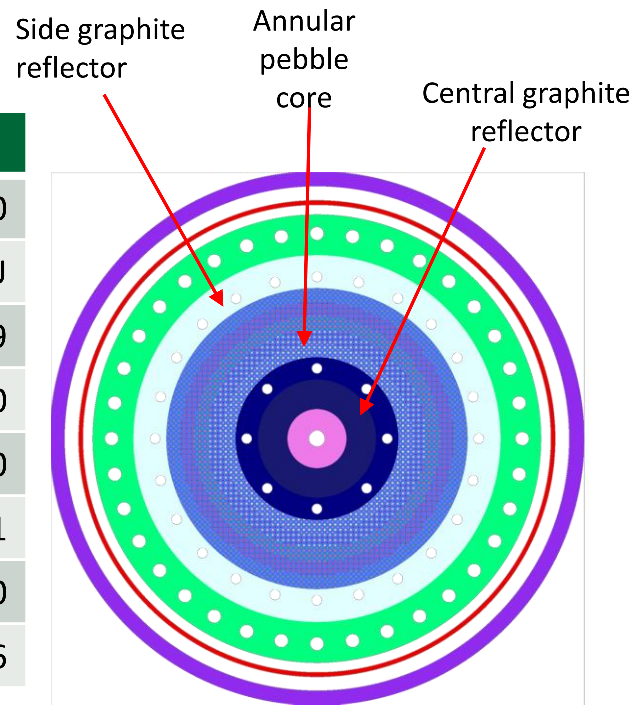
Objective of the study:

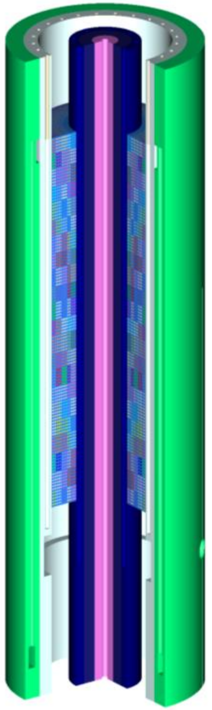
- Assess the changes in fuel burnup, Pu content, and residual ^{235}U enrichment in the discharged pebble due to sensitivities in neutron flux magnitude and spectra, residence time, and initial ^{235}U enrichment.
- Uncertainty estimation to support nuclear material accounting of pebbles stored in used fuel canisters.

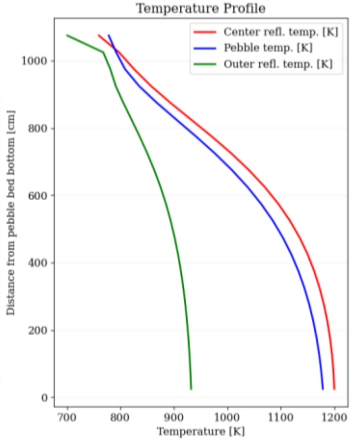
PBMR-400

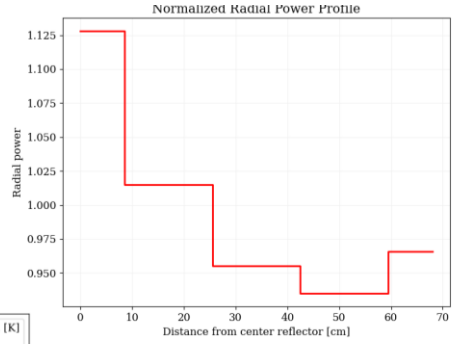
- Developed by a South African company from 1994-2009
- Chosen for this study due to its publicly available reactor design and operation information [1].

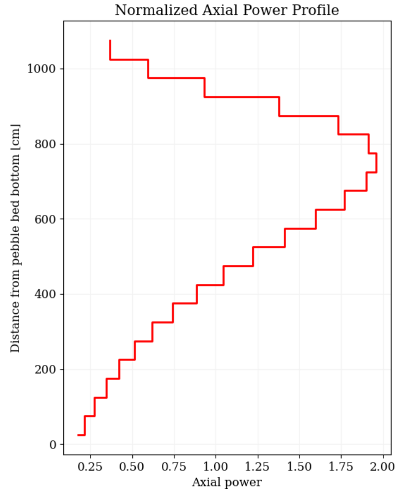
Characteristics	Value
Thermal power (MWth)	400
Fuel type	TRISO, 9.6% ^{235}U
Uranium loading per pebble (g)	9
Number of particles per pebble	15,000
Number of pebbles	~452,000
Specific power (MWth/tU)	98.431
Target discharged burnup GWd/tU	90
Number of passes	6











[1] Nuclear Science Committee, Nuclear Energy Agency (NEA), "PBMR Coupled Neutronics / Thermal Hydraulics Transient Benchmark The PBMR-400 Core Design, vol. 1: The Benchmark Definition," NEA/NSC/DOC(2013)10, Paris, France, 2013.

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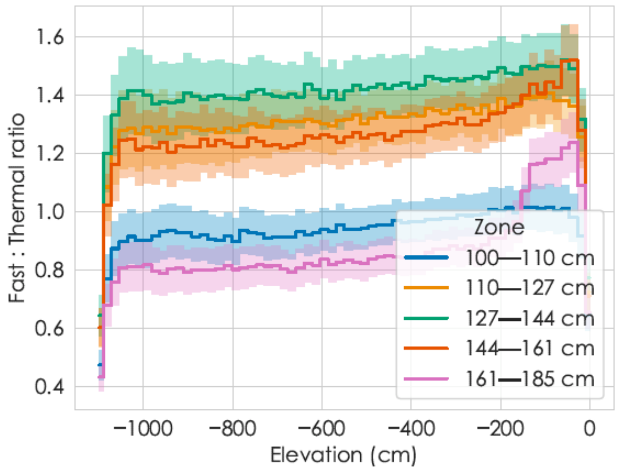
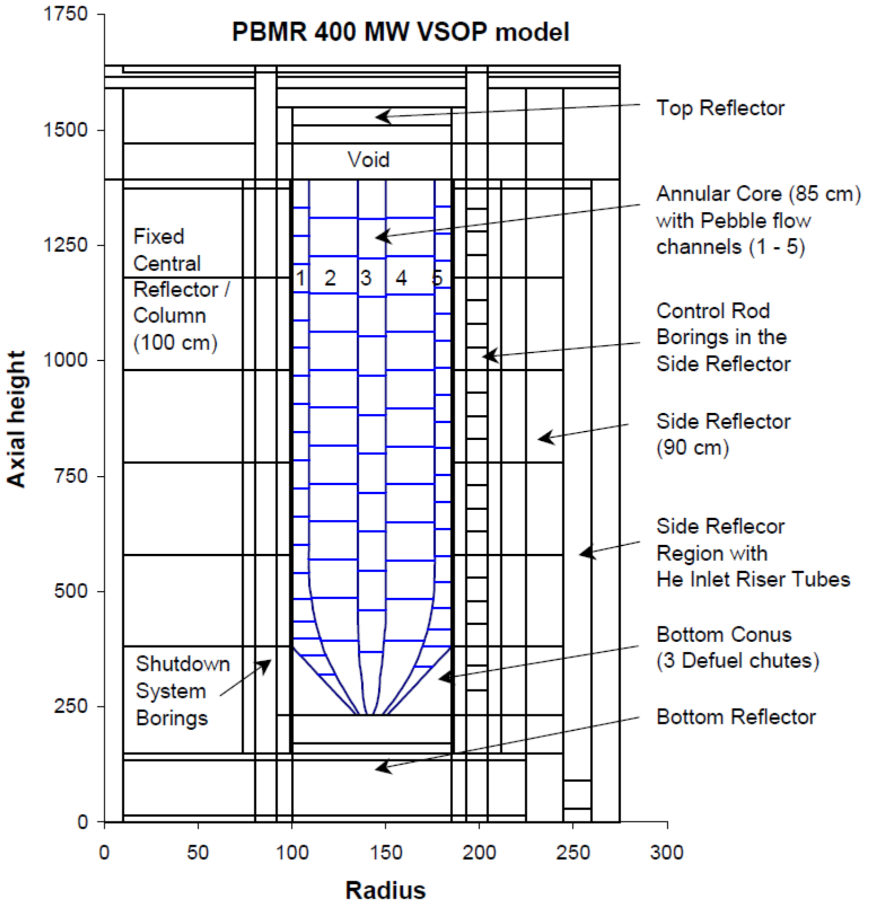
PBMR-400 (Continued)

- Five radial flow channels used to simulate the core.
- Transit time in channels 1 and 5 is 54% longer than the other channels.
- The variations in neutron energy spectra are larger in the radial direction than in the axial direction.
- The neutron spectra are softer in the two channels next to the reflectors than in others.

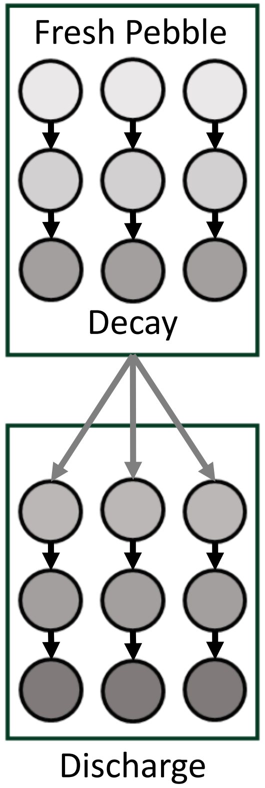




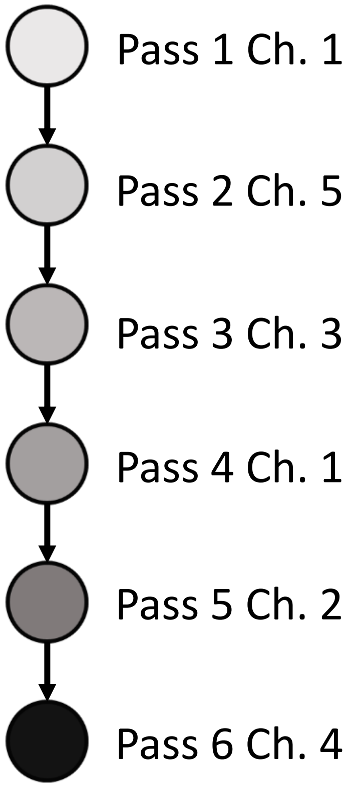
Methodology: SCALE

SCALE/ORIGAMI Capability

- ORIGAMI was recently improved to rapidly model the depletion of flowing pebbles.
- Pebble depletion is carried out as a *series of axial (transit) zones*.
  - The radial characteristics in each transit zone can also be accounted for.
- Simulating multipass:
  - Each pass includes the pebble power, irradiation time, cooling time, and a series of sequential transit zones.
  - The fractional irradiation time and the axial power factor are part of the inputs for each transit zone.



A batch of pebbles in each transit

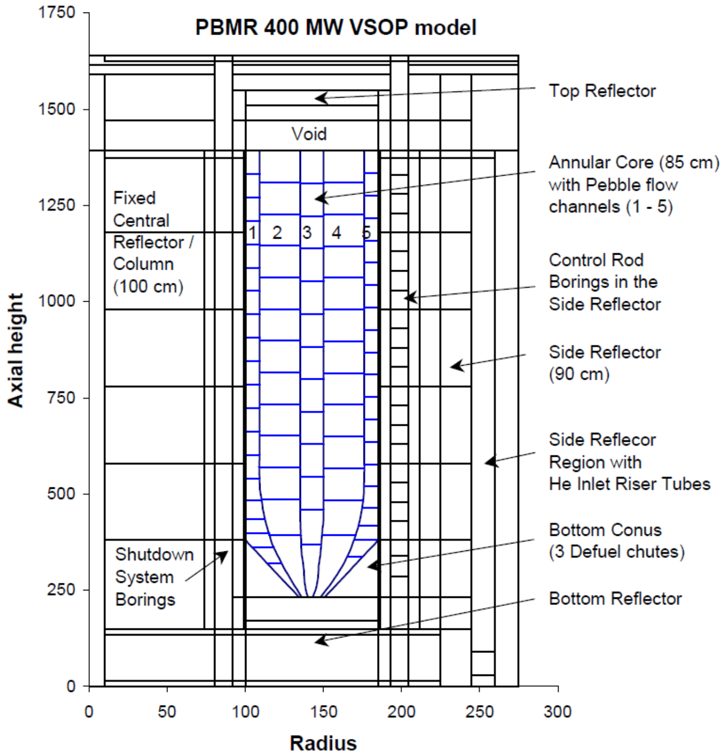


Single pebble transits in random channel in each pass

Steve Skutnik, "Flowing-pebble depletion modeling in ORIGAMI", SCALE Users' Group Workshop, ORNL, April 27-29, 2022.

SCALE Library Generations and Depletion Scheme

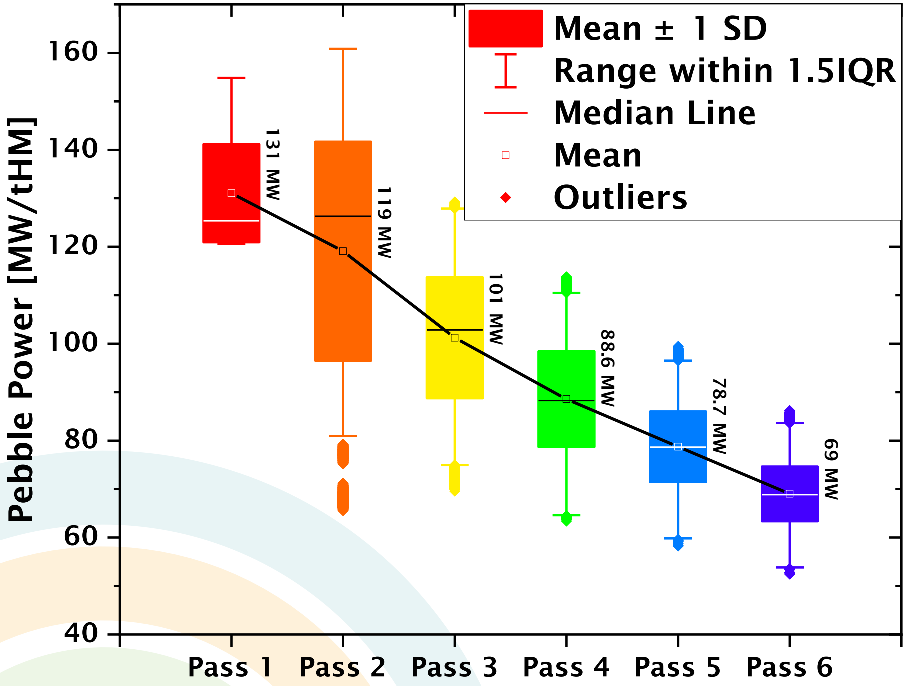
- A series of TRITON models were developed to generate the reactor libraries (in HDF5 format) for the ORIGAMI calculations:
  - 5 radial channels, 22 axial zones, 4.5 d cooling after each pass
  - 3 fuel/reflector temperatures (700, 900, 1200 K)
  - 28 burnup steps (0 to ~100 GWd/MTU)
- At a given point, one of these libraries is called along with user-specified power levels, irradiation/decay time, power peaking factors, etc. to simulate the local/temporal irradiation conditions.
- 20,000 pebbles simulated to account for variations in different paths and passes, power levels, and irradiation time.
- A random channel for each pebble is determined for each pass (based on the channel's volume fraction and velocity).



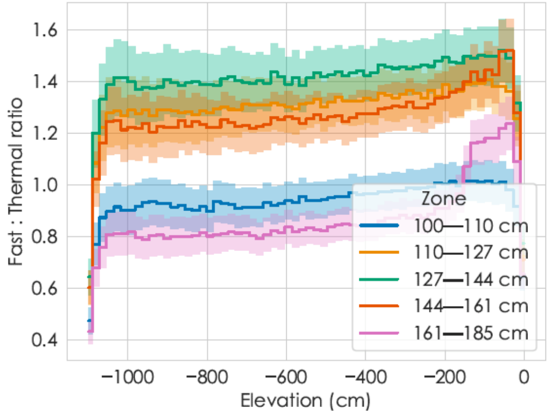
[2] F. REITSMA, "The Pebble Bed Modular Reactor Layout and Neutronics Design of the Equilibrium Cycle," *Proc. PHYSOR 2004*, Chicago, Illinois, USA, April 25–29, 2004, American Nuclear Society (2004) (CD-ROM).

Parameters for Uncertainty Estimation

- A pebble's power history in each pass was obtained from reference [2], with uncertainties added (for the study conducted in this work).



Uncertain Parameter	Range of Uncertainty	Distribution
Radial channel spectrum	Following radial flow probability	1 is fast; 5 is thermal
Transit time	± 2 days	Uniform
Pebble Power	± 2%	Uniform
^{235}U Enrichment	± 0.1%	Uniform
Temperature	± 10K	Uniform

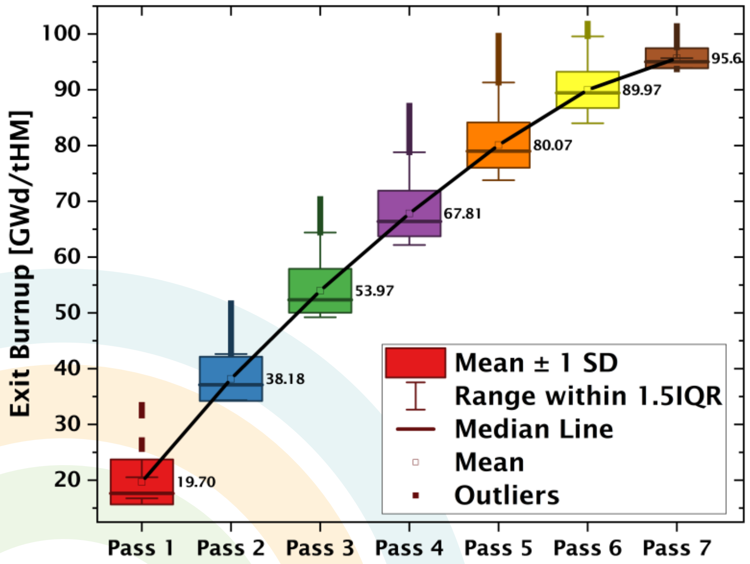


[2] F. REITSMA, "The Pebble Bed Modular Reactor Layout and Neutronics Design of the Equilibrium Cycle," *Proc. PHYSOR 2004*, Chicago, Illinois, USA, April 25–29, 2004, American Nuclear Society (2004) (CD-ROM).

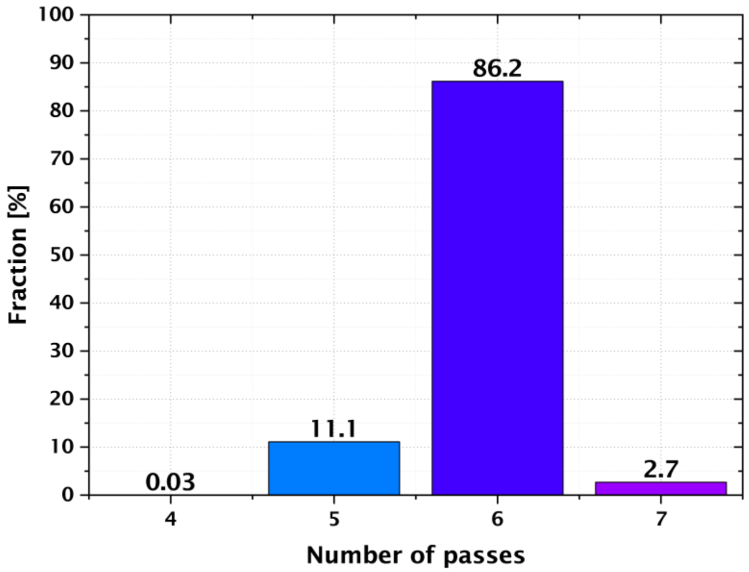
SCALE Simulation Results

Burnup Distribution

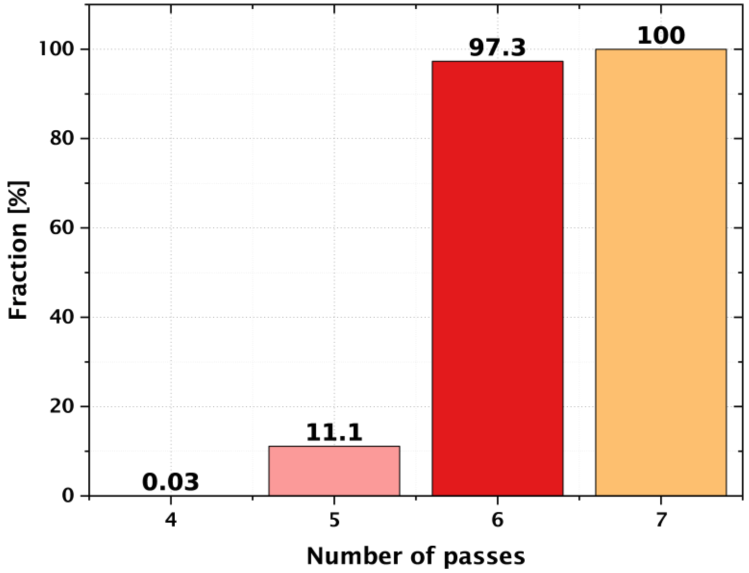
- A pebble can be retired earlier than the target BU since it has a chance to exceed the target if it is returned to the core.
- The BU set point is 85.5 GWd/tU at BUMS.
- The average BU of the retired pebbles is about 90.048 ± 3.266 GWd/tU.



Pebble burnup distribution after each pass



Fraction of permanent discharge pebbles per pass

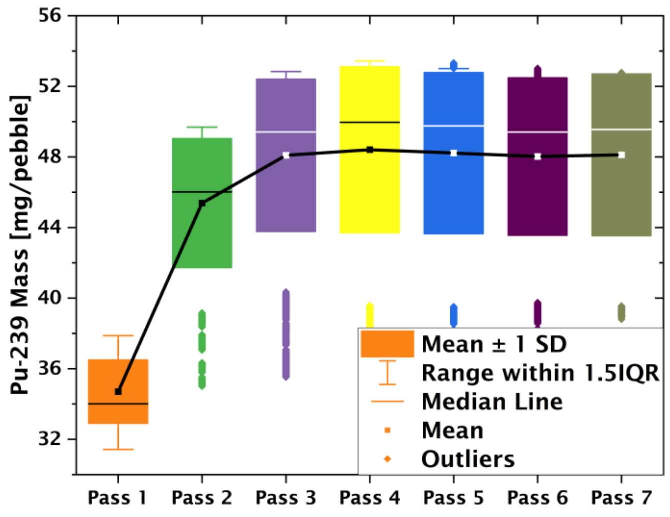


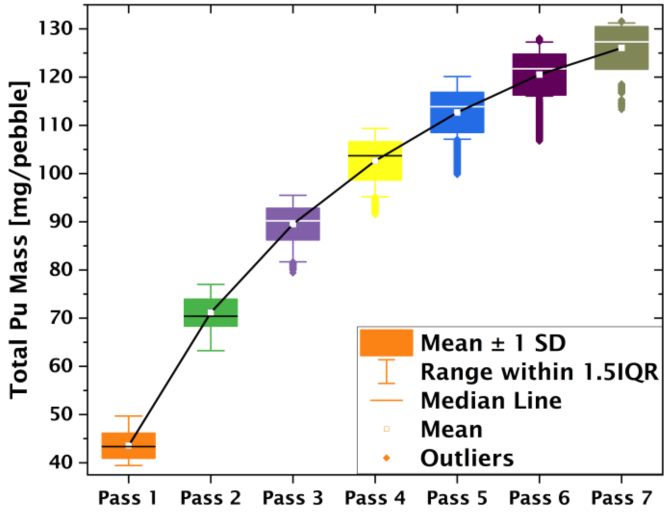
Fraction of pebbles reaching target burnup (85 GWd/tHM) after each pass

SCALE Simulation Results

^{239}Pu and Pu buildup as function of pebble pass

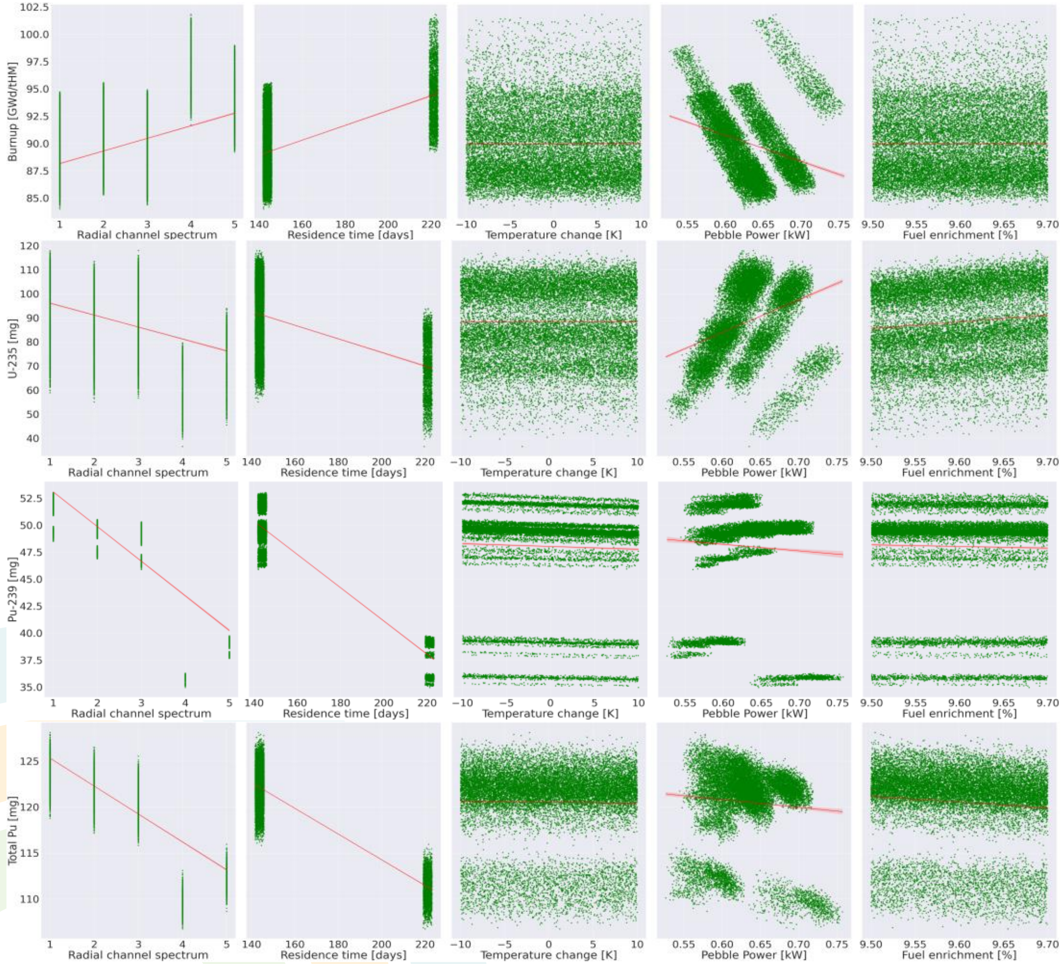
Pass	Avg. ^{239}Pu Mass (mg)	Max. ^{239}Pu Mass (mg)	Avg Pu Mass (mg)	Max. Pu Mass (mg)
1	34.7 ± 1.8	37.9	43.5 ± 2.6	49.7
2	45.4 :± 3.6	49.7	71.2 ± 2.8	76.6
3	48.1 ± 4.3	52.6	89.5 ± 3.3	95.1
4	48.4 ± 4.7	53.2	102.6 ± 4.0	108.8
5	48.2 ± 4.6	53.1	112.7 ± 4.2	119.5
6	48.0 ± 4.5	52.8	120.5 ± 4.2	127.7
7	48.1 ± 4.6	52.5	126.1 ± 4.4	131.7
Retired	47.3 ± 5.0	53.0	119.3 ± 6.0	131.7





SCALE Simulation Results

Pebbles in Pass 6



Pearson Correlation Matrix for Pebbles in Pass 6

	Burnup	U-235 consumption	U-238 consumption	Pu-239 buildup	Total Pu buildup
Radial channel spectrum	0.402	-0.360	-0.247	-0.817	-0.813
Residence time	0.574	-0.497	-0.435	-0.952	-0.909
Pebble power	-0.302	0.358	0.290	-0.056	-0.081
Pebble temp.	0.006	0.003	-0.011	-0.034	-0.015
Fuel enrich.	0.003	0.101	-0.133	-0.020	-0.086

SCALE Simulation Results

^{239}Pu mass change due to nuclear data uncertainty

- The sensitivity coefficient, defined as the relative change in isotope mass (^{239}Pu) resulting from a change in nuclear data, was generated using ORIGEN based on the depletion perturbation theory.
- Can be interpret as:
  - The buildup of ^{239}Pu **increases** by 0.94% if the ^{238}U (n,γ) cross section increases by 1%.
  - The buildup of ^{239}Pu **decreases** by -0.37% if the ^{239}Pu (n,γ) cross section increases by 1%.
  - The buildup of ^{239}Pu **decreases** by -0.60% if the ^{239}Pu (n,f) cross section increases by 1%.

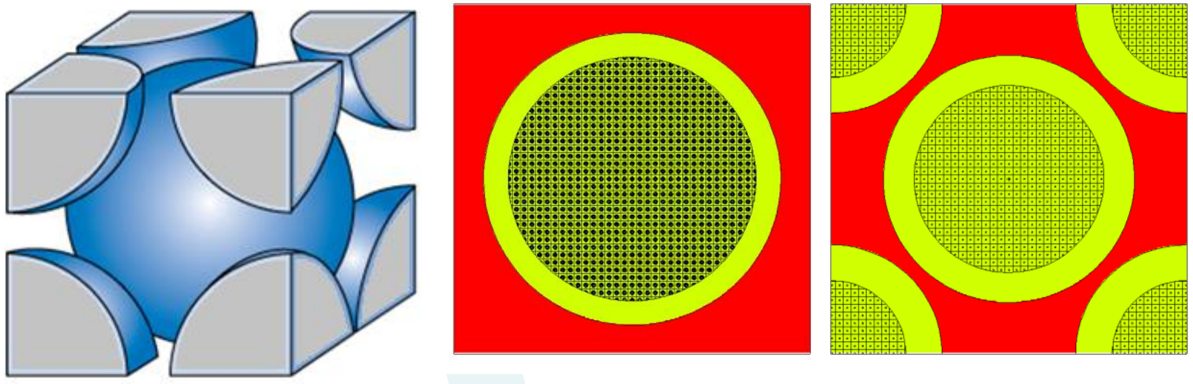
Sensitivity Coefficients of ^{239}Pu in PBMR-400 Pebble

Isotope	Reaction	Radial Channel 1	Radial Channel 2	Radial Channel 3	Radial Channel 4	Radial Channel 5
^{238}U	**(n,γ)**	**9.44E-01**	**9.37E-01**	**9.35E-01**	**9.38E-01**	**9.46E-01**
^{238}Pu	(n,γ)	1.12E-03	8.74E-04	8.15E-04	8.81E-04	1.11E-03
^{236}U	(n,γ)	1.00E-03	7.27E-04	6.67E-04	7.37E-04	1.00E-03
^{235}U	(n,γ)	9.80E-04	7.17E-04	6.60E-04	7.26E-04	9.81E-04
^{237}Np	(n,γ)	8.52E-04	6.29E-04	5.80E-04	6.39E-04	8.57E-04
^{241}Pu	β^-	2.95E-04	2.52E-04	2.39E-04	2.52E-04	2.89E-04
^{240}Pu	(n,γ)	1.90E-04	1.59E-04	1.50E-04	1.61E-04	1.89E-04
^{242}Cm	α	1.66E-04	1.42E-04	1.35E-04	1.42E-04	1.64E-04
^{241}Am	(n,γ)	1.62E-04	1.50E-04	1.46E-04	1.50E-04	1.62E-04
^{238}U	(n,f)	-2.06E-04	-2.28E-04	-2.31E-04	-2.26E-04	-1.80E-04
^{235}U	(n,f)	-2.18E-04	-1.38E-04	-1.22E-04	-1.42E-04	-2.25E-04
^{239}U	β^-	-2.30E-04	-1.85E-04	-1.75E-04	-1.90E-04	-2.38E-04
^{239}Np	(n,γ)	-1.44E-03	-1.34E-03	-1.33E-03	-1.36E-03	-1.49E-03
^{239}Np	β^-	-3.11E-02	-2.48E-02	-2.33E-02	-2.55E-02	-3.21E-02
^{239}Pu	**(n,γ)**	**-3.66E-01**	**-3.67E-01**	**-3.66E-01**	**-3.66E-01**	**-3.64E-01**
^{239}Pu	**(n,f)**	**-6.02E-01**	**-6.00E-01**	**-5.99E-01**	**-6.01E-01**	**-6.03E-01**

MCNP Fuel Burnup Simulation of PBMR-400

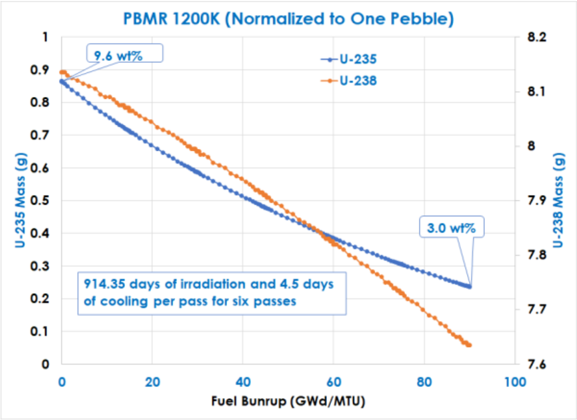
PBMR-400 MCNP Model

- Explicit TRIOS modeling with two pebbles in a BCC
- Packing fraction of 61%
- Power for two pebbles: 1.772 kWth
- 15000 TRISOs/Pebble
- MCNP URAN feature for randomization
- 22 axial locations and six passes

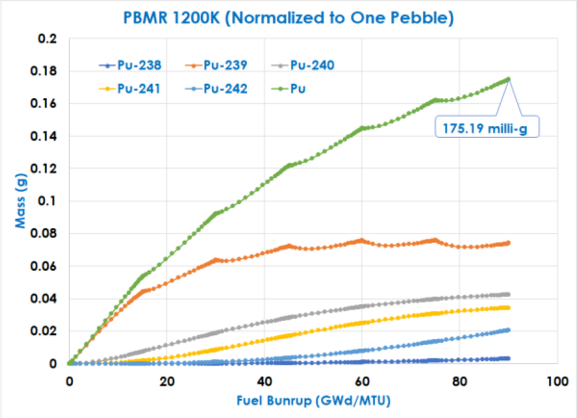


MCNP Model Visualization

Power (kWth) in each axial location; residence time 6.9269 d per location
0.6572
1.0742
1.6724
2.4480
3.0583
3.3824
3.4619
3.3564
3.1265
2.8251
2.4937
2.1619
1.8487
1.5644
1.3128
1.0941
0.9059
0.7447
0.6064
0.4868
0.3828
0.3140



Uranium depletion



Plutonium buildup

MCNP Simulation Results Summary

Parameter Perturbed	k-inf	Burnup (GWd/MTU)	Pu content (milli-g) in a Pebble	Residual ^{235}U wt% in a Pebble
Temperature 900K (100% Power)	1.00057	89.96	169.82	2.6
Temperature 1200K (100% Power)	0.98394 (0.97150)*	89.96 (89.92)	175.19 (179.92)	3.0 (3.0)
Temperature 1200K (102% Power)	0.97993	91.76	176.58	2.9
Temperature 1200K (98% Power)	0.99010	88.16	173.79	3.1
Temperature 1200K (100% Power) 2 days additional transit time/pass	0.98188	91.14	176.11	3.0
Temperature 1200K (100% Power) 2 days lower transit time/pass	0.98816	88.78	174.33	3.1
Temperature 1200K (100% Power) 9.7 ^{235}U wt% (0.1 % higher)	0.98694	89.96	175.69	3.1
Temperature 1200K (100% Power) 9.5 ^{235}U wt% (0.1 % lower)	0.98163	89.96	174.89	2.9
Average	0.98664	89.96	174.55	3.0
% Standard Deviation	0.7 %	1.3 %	1.2 %	5.7 %

SCALE Result point check with MCNP

- The most sensitive parameters to fuel burnup and Pu production are the residence time and the neutron spectrum in the channel.
- These two parameters are applied in the simulations by grouping the pebbles into several radial and axial flow channels, stressing the need for an accurate definition of the radial and axial flow channels.
- These definitions can be guided either using experiments or discrete element modeling and simulation.

Conclusions

- SCALE and MCNP simulations performed to estimate fuel burnup and fissile content sensitivities to neutron flux and neutron spectra, residence time, temperature, and initial ^{235}U enrichment.
- MCNP and SCALE simulations agree very well when a point check was performed.
- Detailed SCALE simulations on various parametric sensitivities showed a maximum variation of about ±10 mg Pu per pebble, considering one-sigma standard deviation.
- MCNP results were found to be conservative from an NMAC perspective for the simulations performed using average initial ^{235}U enrichment and other reactor parameters (temperature, residence time, neutron flux, neutron spectrum) compared to SCALE simulation results.
  - 175 mg (MCNP) vs 125 mg (SCALE).
- These differences due to variability in operational parameters and the resulting fuel burnup and fissile content in pebbles manifest themselves to impact NMAC/Safeguards, specifically in Shipper/Receiver Difference (SRD) or declared value vs verification measurement value.
  - Can also impact radiation dose calculations.

Thank you!

Questions?