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Neutronic Evaluation of a SMR using the MCNP and SERPENT Codes

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Introduction

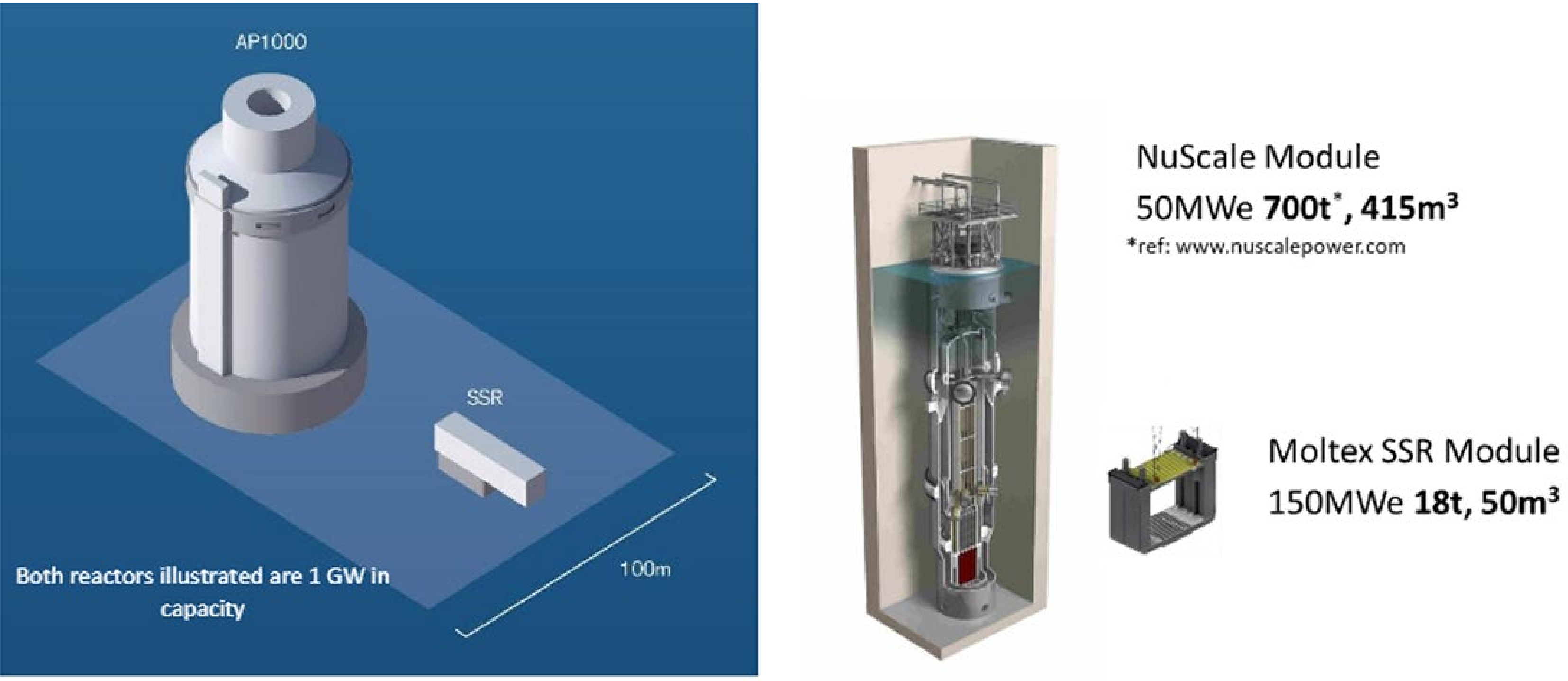
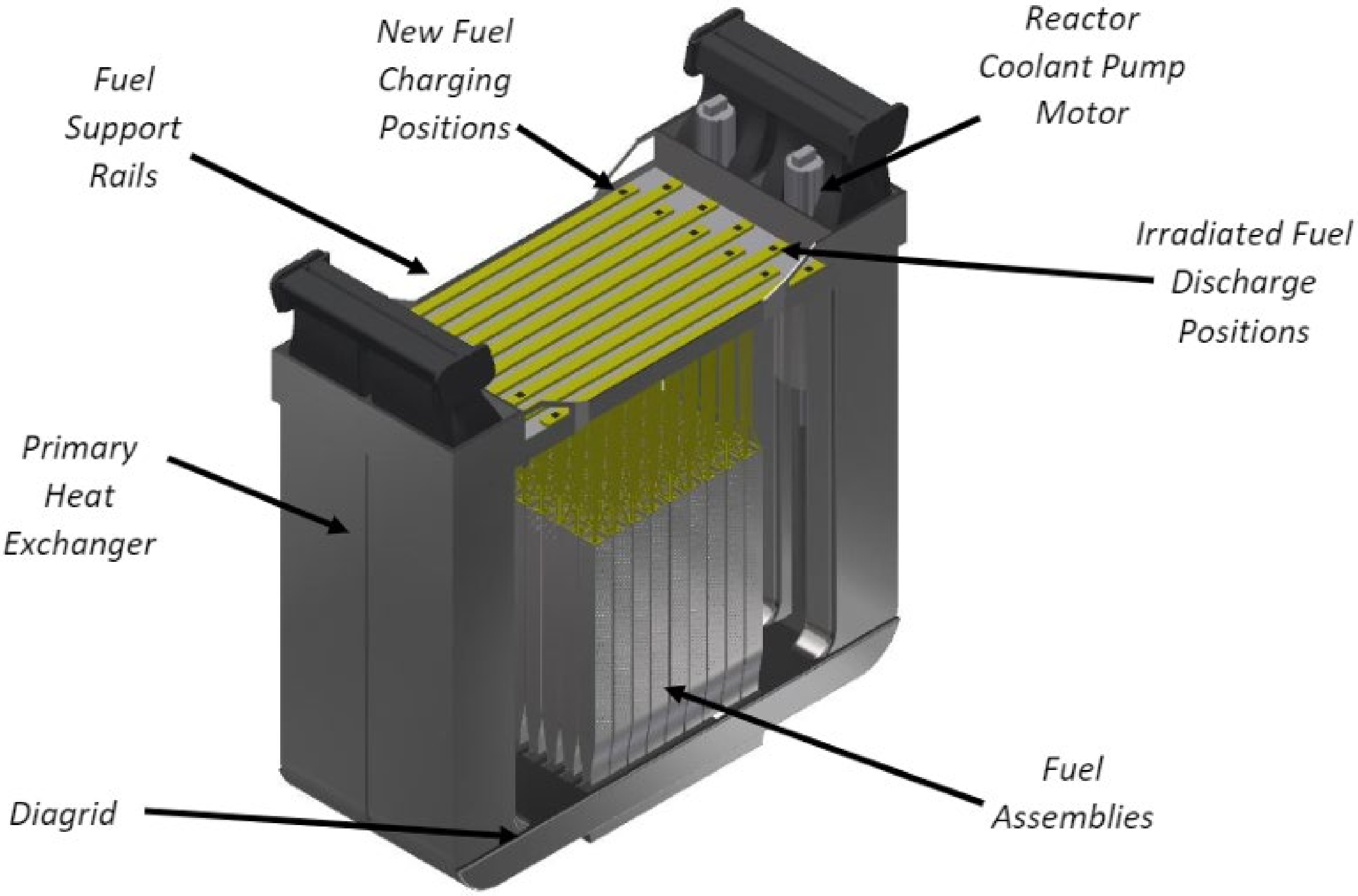


Figure 1: To-scale comparisons of the SSR-W1000 with AP1000 (left) and an SSR-W module with a a NuScale module.

Small modular reactors (SMRs) are advanced nuclear reactors that have a power capacity of up to 300 MW(e) per unit, which is about one-third of the generating capacity of traditional nuclear power reactors. Their main advantages are:

- ❖ Physically a fraction of the size of a conventional nuclear power reactor.
- ❖ Its modular characteristic, making it possible for systems and components to be factory-assembled and transported as a unit to a location for installation.
- ❖ Offer savings in cost and construction time, and they can be deployed incrementally to match increasing energy demand

Introduction



Overview of Stable Salt Reactor Core Module

The principle module parameters are:

- Core Power
  - 375 MW(th)
  - 150 MW(e)
- Salt nominal temperatures
  - Fuel salt = 760 °C
  - Coolant salt = 588 °C
- Layout
  - 10x10 array of fuel assemblies
  - 18x18 fuel rods
- Dimensions – Indidual Module Core
  - 2.05 m length across fuel assemblies
  - 2.05 m width across fuel assemblies
  - 3.7 m height of fuel (incl. Support frames)
  - 1.6 m active height of fuel assemblies
- Dimensions – Reactor Tank External for 300 MW(e)
  - 6 m length (along modules)
  - 5.3 m length (across modules)
  - 4.2 m overall height

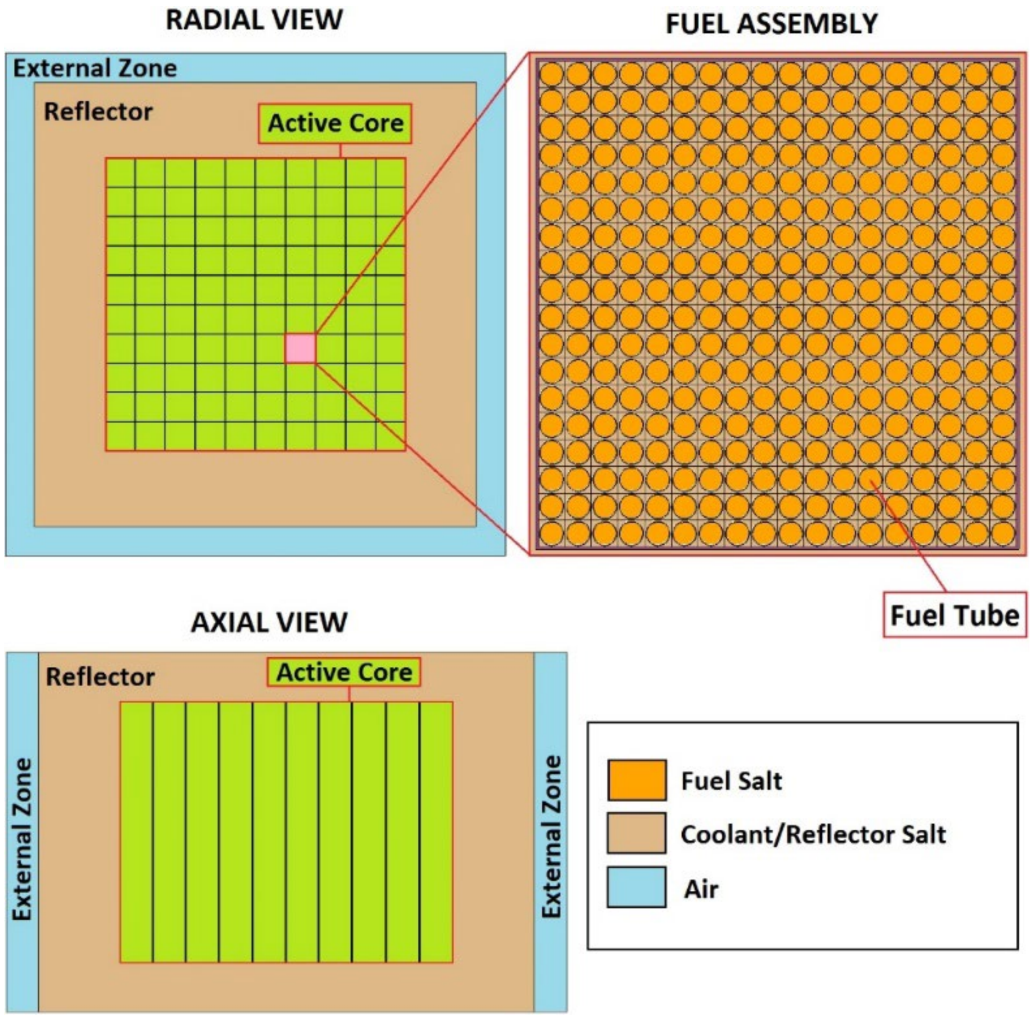
Objective

This work aims to assess the neutronic behavior in steady state and later during burnup, for the SSR-W core, using the Monte Carlo codes MCNP6.2 [3] and SERPENT 2.1.32 [4]. The reactor was simulated for full power temperatures of the components using data from the ENDF/B-VIII.0 cross-section libraries [5] and the NJOY99.396 code [6].

Methodology

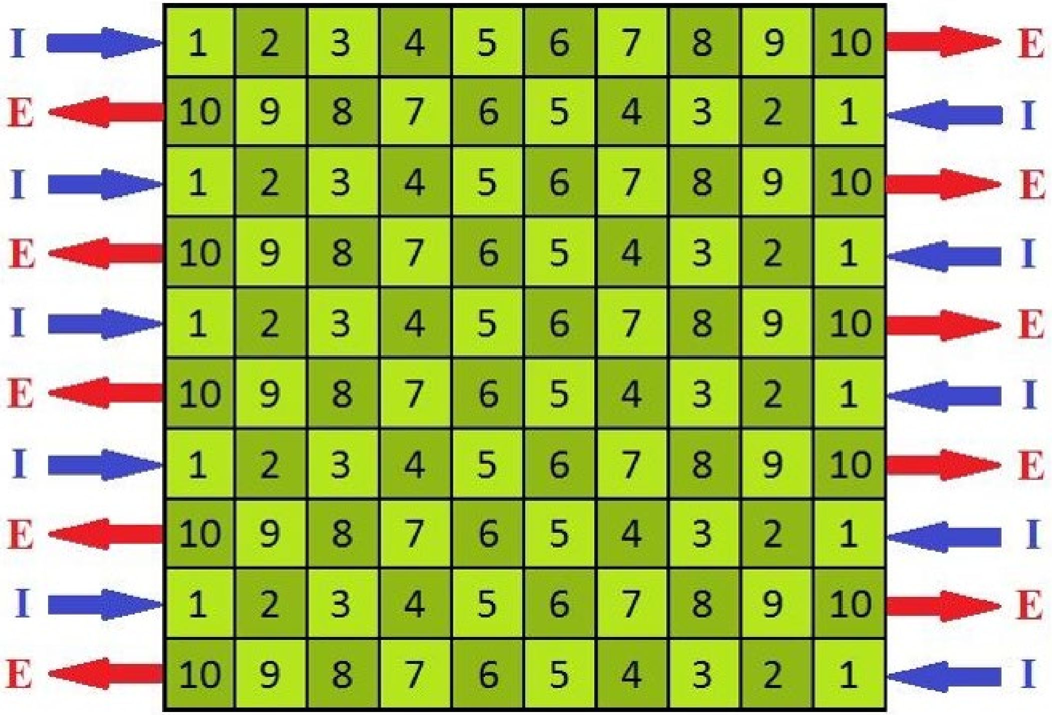
The SSR-W core was modeled with the MCNP6.2 and SERPENT 2.1.32 codes, using the ENDF/B-VIII.0 neutrons cross-section sublibrary.

Core Parameters	
# assemblies	100
# rods per assemblies	324
Core Power (MWth)	375
Fuel salt composition	UCl_3 (20%) / $PuCl_3$ (20%) / NaCl (60%)
Fuel salt temperature (°C)	760
Fuel salt density (g/cm^3)	3.2379
Coolant salt composition	ZrF_4 (39.2%) - ZrF_2 (1.96%) - NaF (10%) - KF (48%) - HfF_4 (0.8%) - HfF_2 (0.04%)
Coolant salt temperature (°C)	588
Coolant salt density (g/cm^3)	2.6603
Cladding	HT9
Reflector	ZrF_4 (39.2%) - ZrF_2 (1.96%) - NaF (10%) - KF (48%) - HfF_4 (0.8%) - HfF_2 (0.04%)



Methodology

In an SSR, the fuel is burned over a period of 5 years. Every six months, the fuel elements are moved horizontally to allow fresh fuel to enter.



Fuel progress across the core (I: inlet fuel, E: outlet fuel).

Fuel Salt		Coolant Salt	
Isotope	Atomic Fraction	Isotope	Atomic Fraction
^{238}Pu	2.8780E-03	^{90}Zr	6.6709E-02
^{239}Pu	3.5648E-02	^{91}Zr	1.4387E-02
^{240}Pu	1.7812E-02	^{92}Zr	2.1752E-02
^{241}Pu	8.8003E-03	^{94}Zr	2.1574E-02
^{242}Pu	6.2902E-03	^{96}Zr	3.4032E-03
^{233}U	---	^{19}F	6.8944E-01
^{234}U	---	^{23}Na	3.1056E-02
^{235}U	1.4468E-04	^{39}K	1.3948E-01
^{236}U	---	^{40}K	1.7061E-05
^{238}U	7.1284E-02	^{41}K	9.5747E-03
^{35}Cl	4.9349E-01	^{174}Hf	4.2831E-06
^{37}Cl	1.4937E-01	^{176}Hf	1.3921E-04
^{23}Na	2.1429E-01	^{177}Hf	4.8946E-04

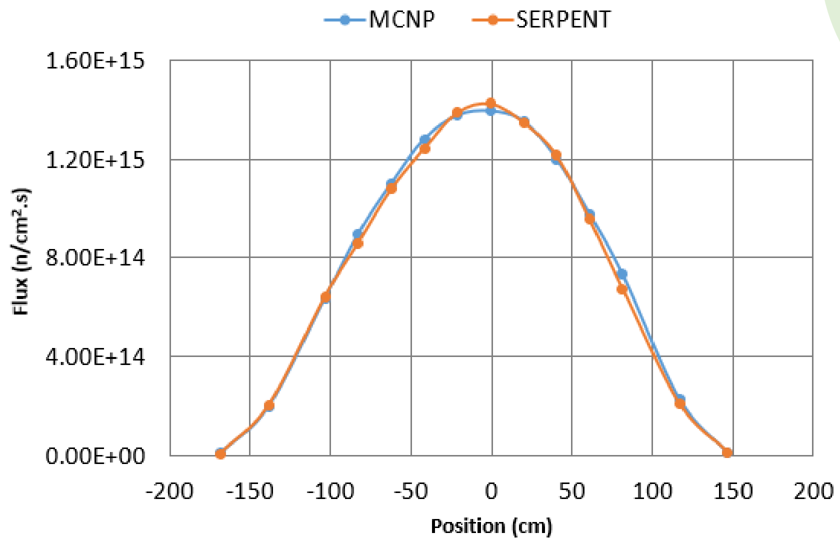
Fuel salt with 3.1% of fissile material obtained from PUREX-type reprocessing.

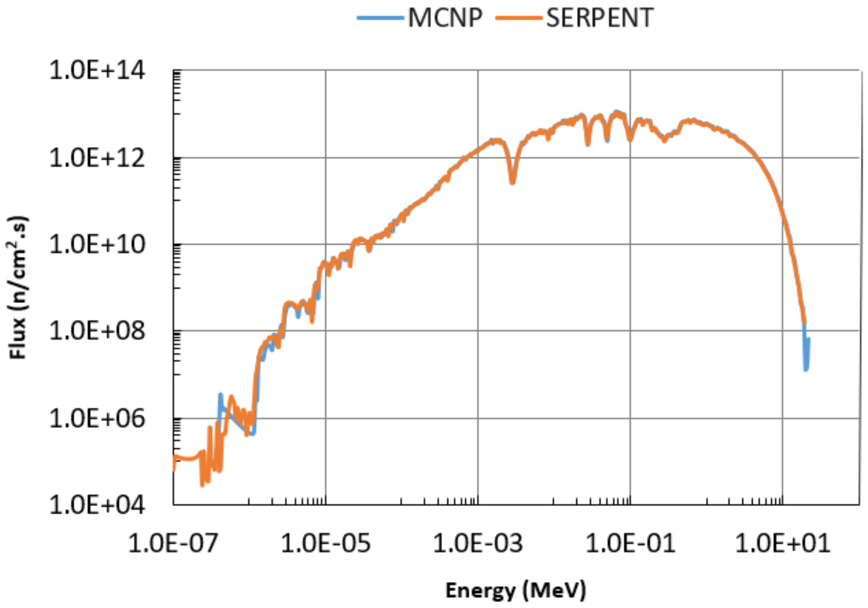
Results

Parameters obtained for steady state, simulated with 10,000 particles for 400 cycles, with 100 inactive cycles in both codes.

Parameters	MCNP	SERPENT	Code Differences
k_{eff}	1.16123 ± 41 pcm	1.16157 ± 23 pcm	-34.00 pcm
β_{eff}	0.00311	0.00325	-4.40 %
P_{NL}	0.99050	0.99016	0.03 %
α_{TF}^{+} (pcm/K)	-8.10647	-8.52078	-5.11 %
α_{TF}^{-} (pcm/K)	-8.94140	-8.70144	2.68 %
f	0.10628	0.10696	0.64 %
ϕ_{RS} (n/cm^2.s)	7.5386E+12	7.0500E+12	-6.48 %
ϕ_{core} (n/cm^2.s)	8.5822E+14	8.5725E+14	-0.11%

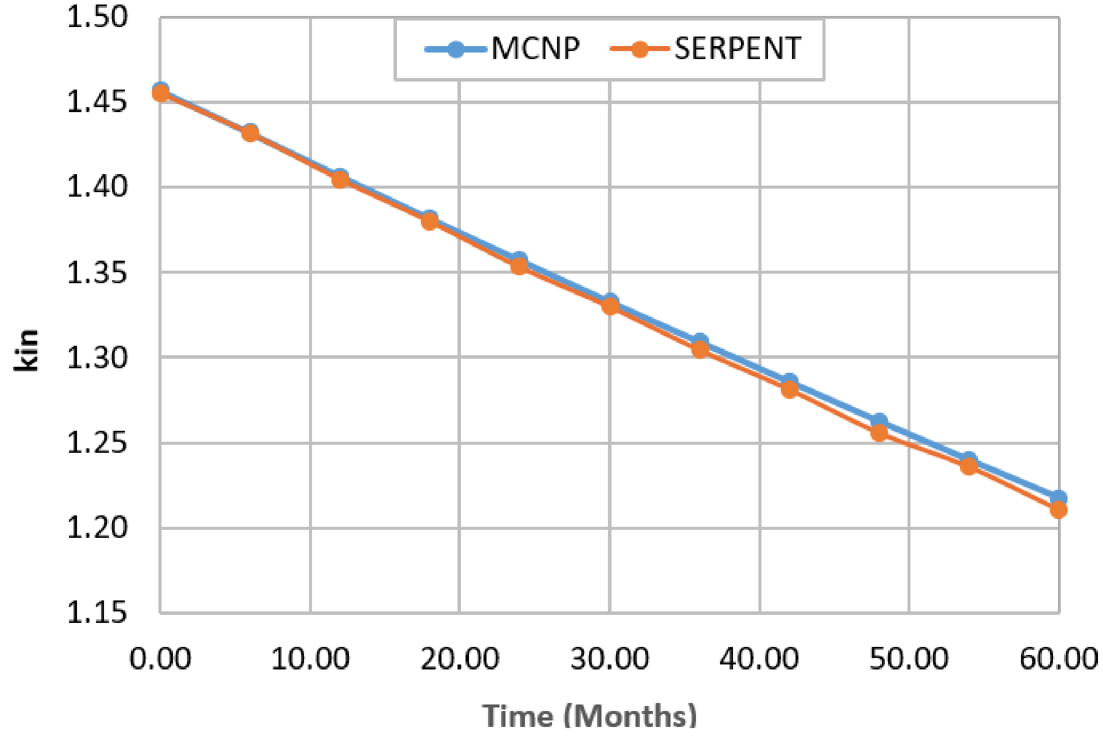
f corresponding to returning neutrons fraction
ϕ_{RS} is the reflector surface flux

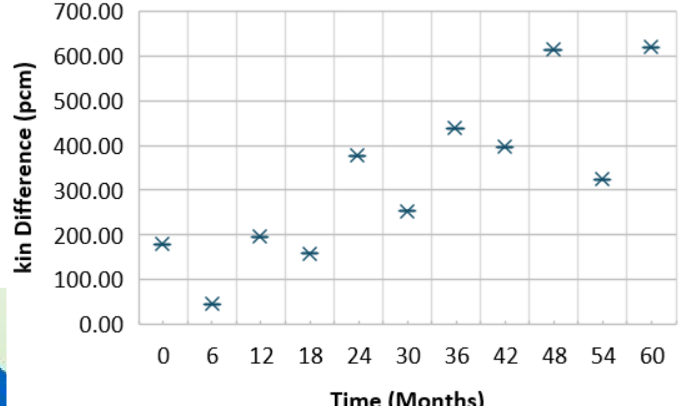




Results

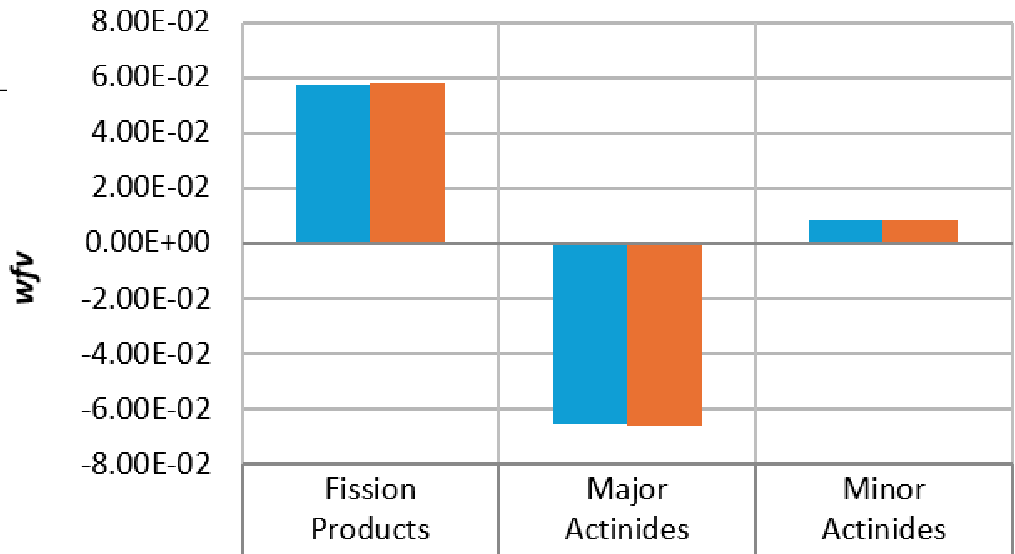
The burnup simulation in both codes was carried out only for one assembly, with an equivalent power of 3.75 MW(th), burnup intervals of 6 months, and a total burnup of 5 years.



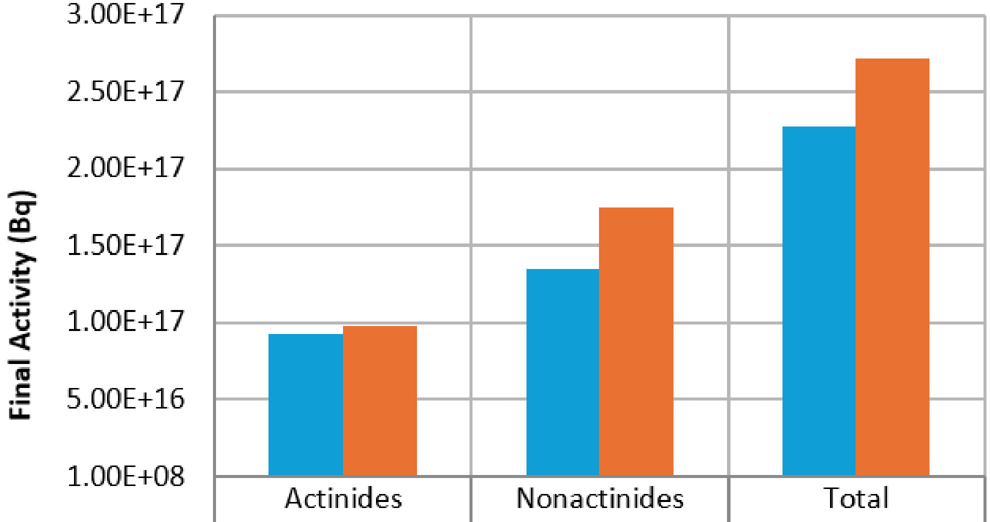


FP Specific Activity (Bq/kg)

MCNP	SERPENT
Ru-105	Ag-118m
2.489E+20	1.770E+24
I-135	Ag-114
1.308E+20	7.966E+23
Xe-135	Rh-104
9.402E+19	9.497E+22
I-130	In-114
7.222E+19	5.097E+22
Eu-157	Ag-117
4.869E+19	4.904E+22
Pr-142	Ag-108
4.274E+19	2.707E+22



	Fission Products	Major Actinides	Minor Actinides
MCNP	5.71E-02	-6.54E-02	7.94E-03
SERPENT	5.79E-02	-6.58E-02	7.98E-03



	Actinides	Nonactinides	Total
MCNP	9.22E+16	1.35E+17	2.27E+17
SERPENT	9.70E+16	1.75E+17	2.72E+17

Results

Begin of Cycle

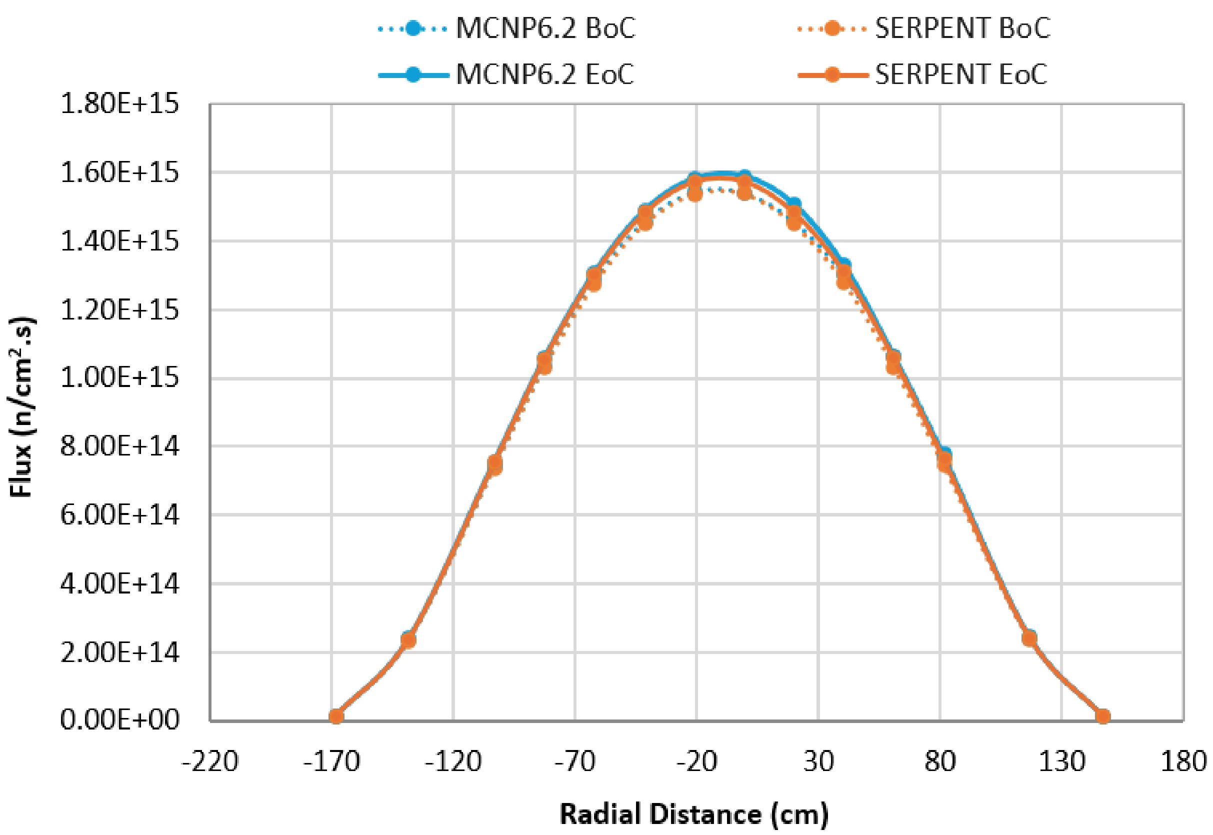
0	6	12	18	24	30	36	42	48	54
54	48	42	36	30	24	18	12	6	0
0	6	12	18	24	30	36	42	48	54
54	48	42	36	30	24	18	12	6	0
0	6	12	18	24	30	36	42	48	54
54	48	42	36	30	24	18	12	6	0
0	6	12	18	24	30	36	42	48	54
54	48	42	36	30	24	18	12	6	0
0	6	12	18	24	30	36	42	48	54
54	48	42	36	30	24	18	12	6	0

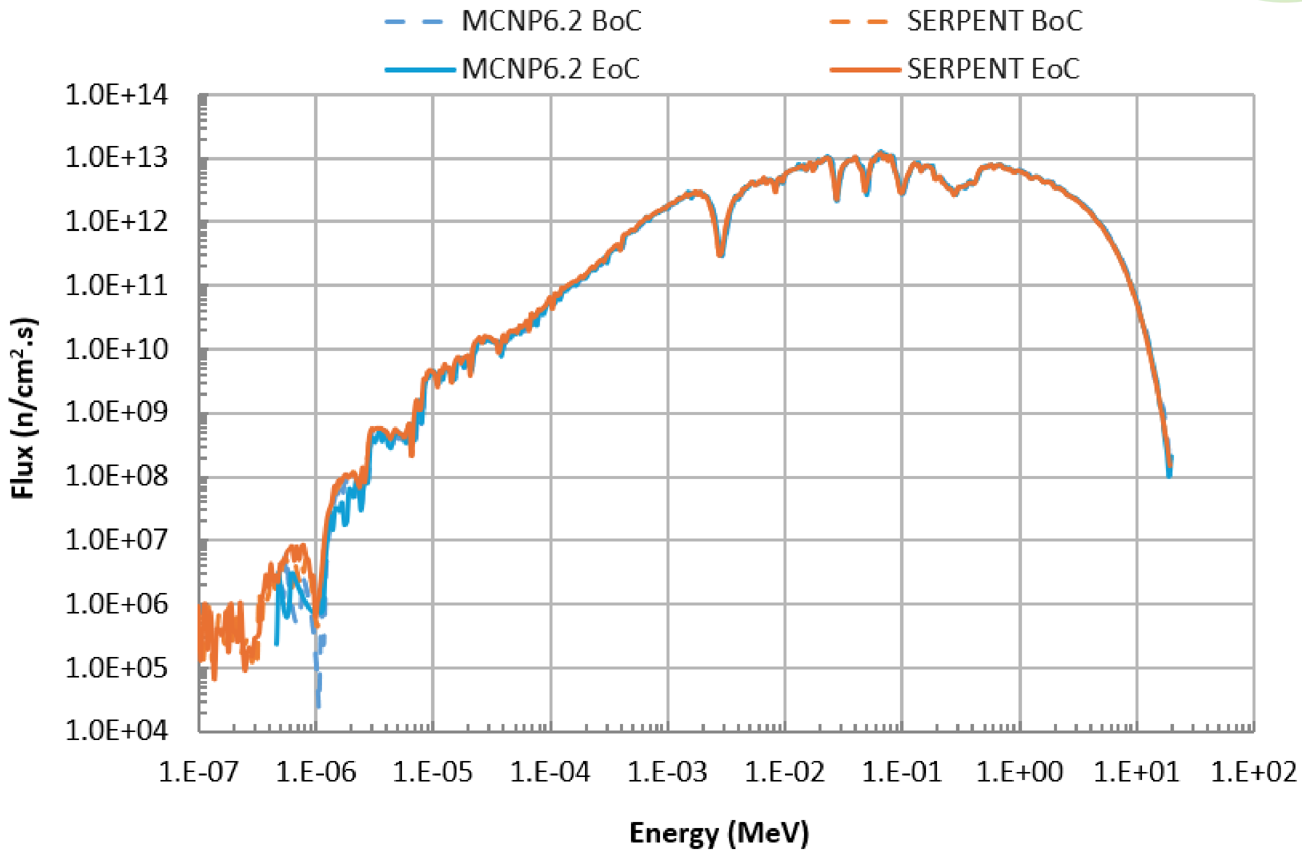
End of Cycle

6	12	18	24	30	36	42	48	54	60
60	54	48	42	36	30	24	18	12	6
6	12	18	24	30	36	42	48	54	60
60	54	48	42	36	30	24	18	12	6
6	12	18	24	30	36	42	48	54	60
60	54	48	42	36	30	24	18	12	6
6	12	18	24	30	36	42	48	54	60
60	54	48	42	36	30	24	18	12	6
6	12	18	24	30	36	42	48	54	60
60	54	48	42	36	30	24	18	12	6

	Begin of Cycle			End of Cycle		
	MCNP	SERPENT	Code Differences	MCNP	SERPENT	Code Differences
Fissile (kg)	1839.89	1839.64	-0.01%	1786.96	1786.61	-0.02%
Fissionables (kg)	4522.50	4522.90	0.01%	4505.33	4505.87	0.01%
Fission Products (kg)	311.89	315.04	1.01%	381.01	385.02	1.05%
Total (kg)	12093.80	12097.01	0.03%	12092.50	12096.67	0.03%
k_{eff}	1.06951	1.06898	-53.00 pcm	1.04992	1.04933	-59.00 pcm
β_{eff}	0.00309	0.00322	3.95%	0.00322	0.00317	-1.65%
P_{NL}	0.99127	0.99061	-0.07%	0.99133	0.99056	-0.08%
f	0.10725	0.10680	-0.41%	0.10610	0.10665	0.51%
ϕ_{RS} (n/cm^2.s)	8.2873E+12	7.7338E+12	-6.68%	8.4466E+12	7.8993e+12	-6.48%
ϕ_{core} (n/cm^2.s)	9.5463E+14	9.5076E+14	-0.41%	9.7840E+14	9.7273E+14	-0.58%

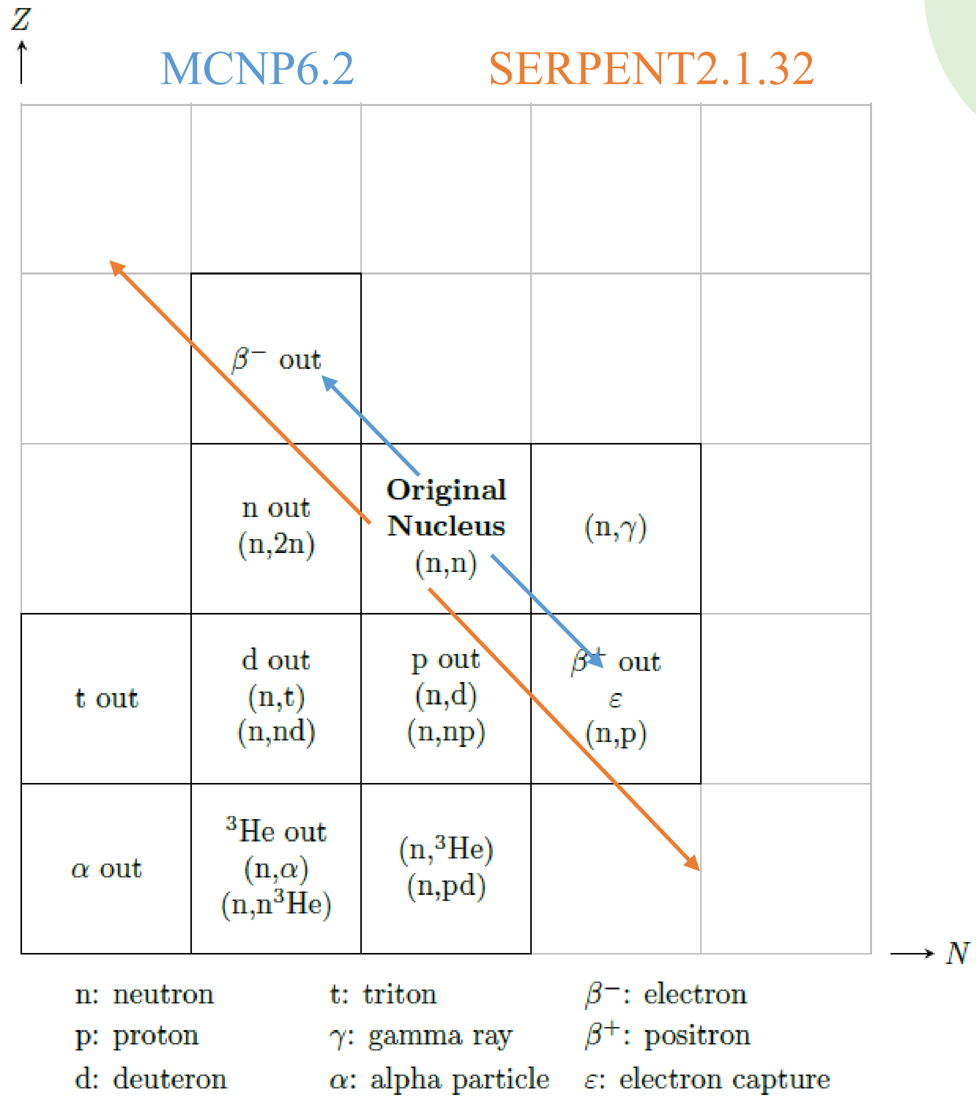
Results


MCNP6.2 BoC
SERPENT BoC
MCNP6.2 EoC
SERPENT EoC
Flux (n/cm².s)
1.80E+15
1.60E+15
1.40E+15
1.20E+15
1.00E+15
8.00E+14
6.00E+14
4.00E+14
2.00E+14
0.00E+00
-220
-170
-120
-70
-20
30
80
130
180
Radial Distance (cm)


MCNP6.2 BoC
SERPENT BoC
MCNP6.2 EoC
SERPENT EoC
Flux (n/cm².s)
1.0E+14
1.0E+13
1.0E+12
1.0E+11
1.0E+10
1.0E+09
1.0E+08
1.0E+07
1.0E+06
1.0E+05
1.0E+04
1.E-07
1.E-06
1.E-05
1.E-04
1.E-03
1.E-02
1.E-01
1.E+00
1.E+01
1.E+02
Energy (MeV)

Conclusion

- ✓ Although the mass variation between the codes presents very similar values, referring to the consumption and production of actinides and fission products, respectively, the final activity of the burnup inventory presents a percentage difference of around 16%.
- ✓ The CINDER90 library file used for burnup in MCNP6.2 can only track a limited number of isotopes in its decay chain algorithm.
- ✓ In principle, in SERPENT 2.1.32, there is no such limitation, and it is also possible to work with updated decay libraries, such as ENDF/B-VIII.0, used in this work.



Nuclides selected for inclusion by the Isotope Generator Algorithm

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