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Previous analysis of fuels transuranic inserted into the hybrid fusion-fission ARC reactor

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INTRODUCTION

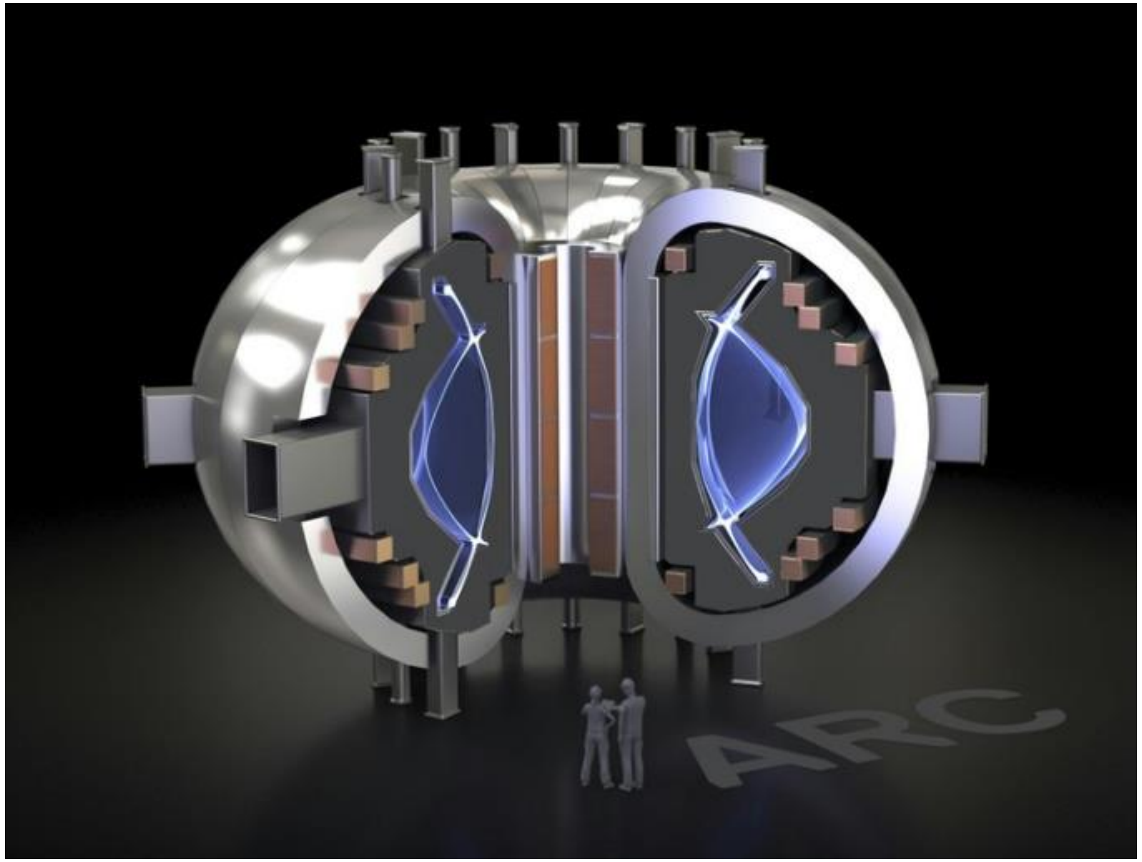

Figure 1 - Illustration of the reference Affordable Robust Compact Reactor (ARC) [1].

Table I - Geometric parameters of the ARC system [1].

Parameters	**ARC**
Major radius, R_0 (m)	3.3
Minor radius, r (m)	1.13
Fusion power (MW)	525
Plasma elongation, k	1.84
Toroidal magnetic field (T)	9.2
Plasma current, I_p (MA)	7.8
Type of plasma	D-T
Volume plasma chamber (m^3)	141

INTRODUCTION

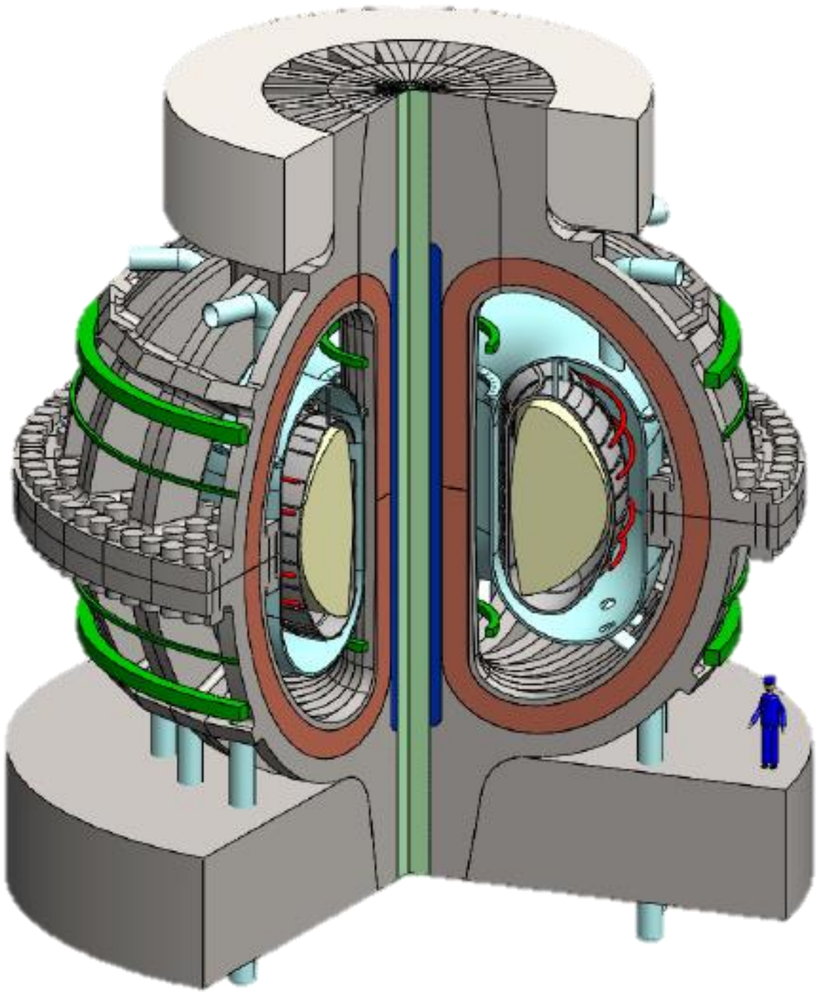

Figure 2 - Illustration of the configuration of the reactor design ARC [1].

Figure 3 - High-Temperature Superconductors (HTS) [1].

INTRODUCTION

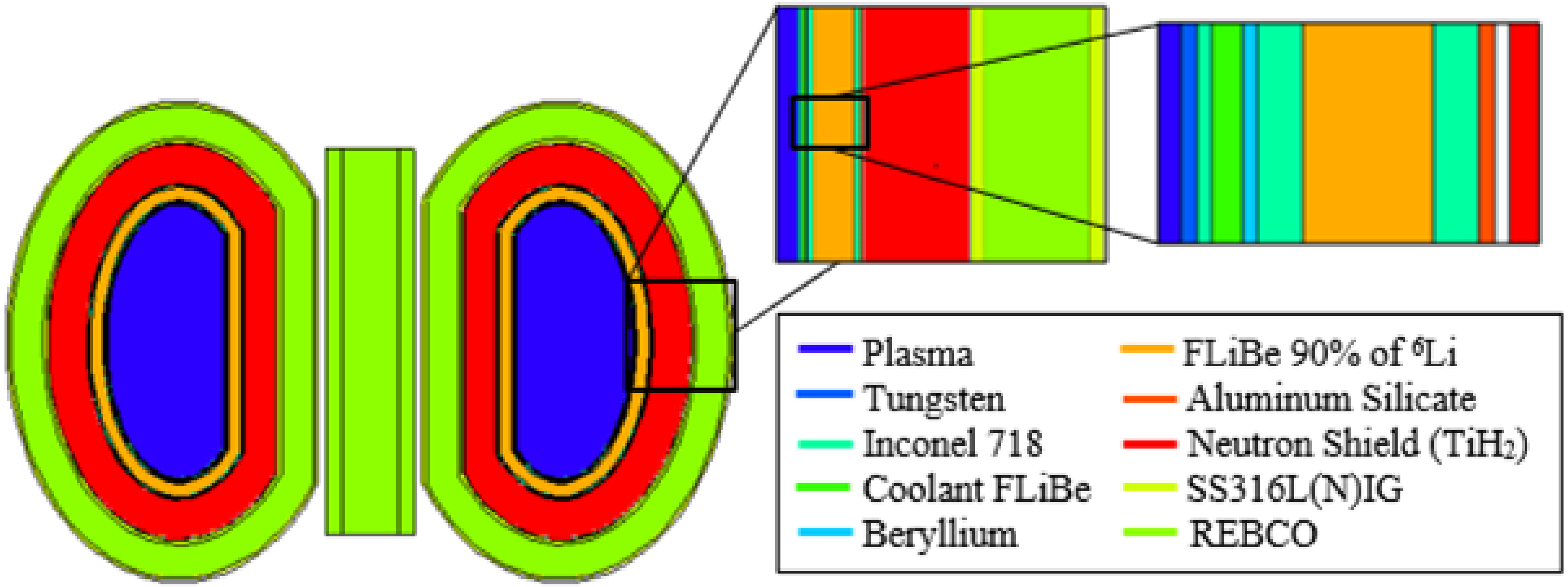

Figure 4 - Simplified model D-shape format design of the ARC reactor (view in the XZ plane) [2].

INTRODUCTION

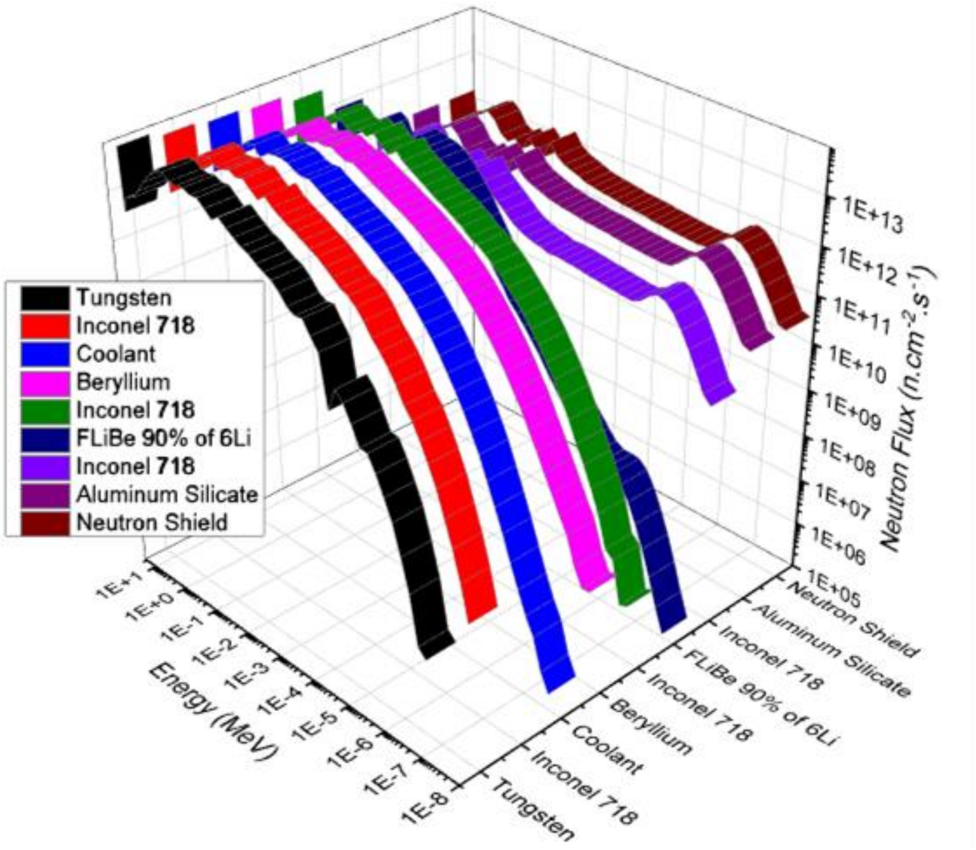

Figure 5 - Neutron flux in the ARC system [2]

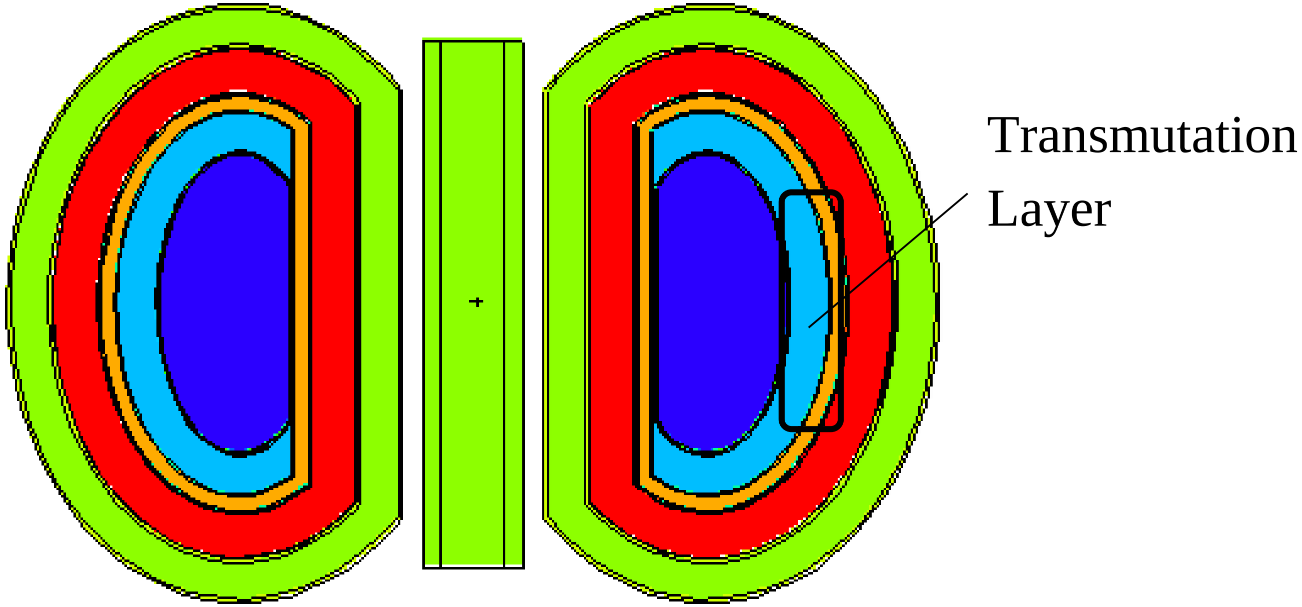

Figure 6 - Simplified model D-shape of the ARC reactor with fission transmutation layer (view in the XZ plane) [2].

METHODOLOGY

Simulations used the Monte Carlo N-Particle code (MCNP5)

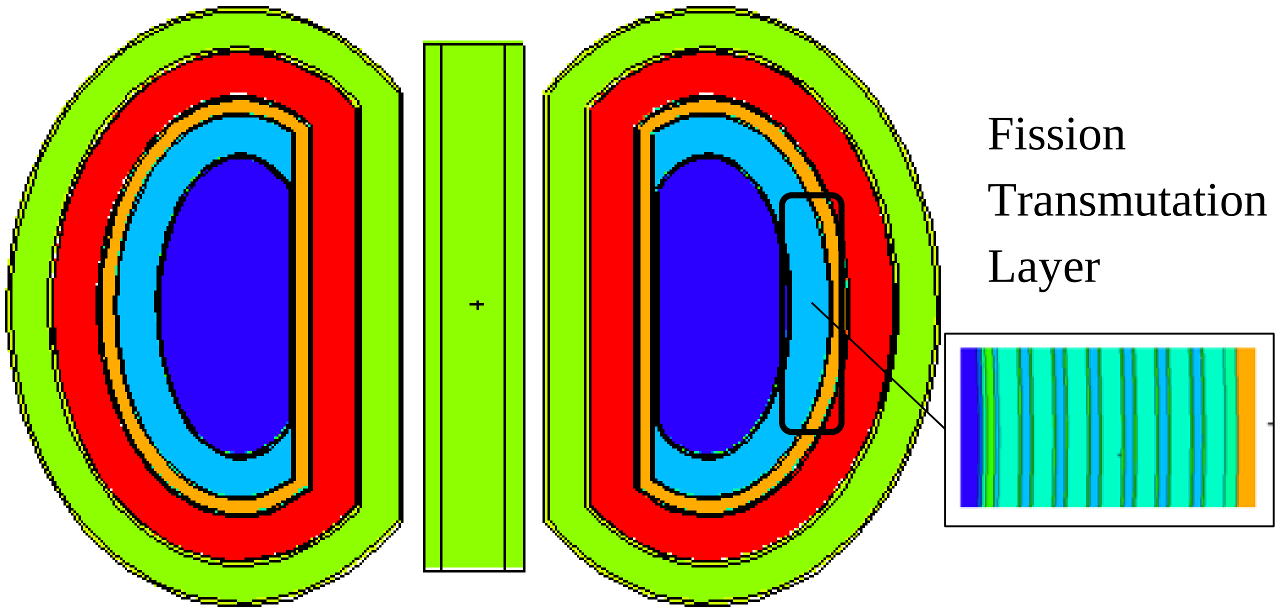

Figure 7 - Simplified model D-shape of the ARC reactor with fission transmutation layer (view in the XZ plane) [2].

Table II - Composition of the transmutation layer [2].

Components	Volume (cm^3)
Fuel	5.2177×10^6
Cladding HT-9	2.6459×10^6
Coolant LBE	1.8597×10^7
Total volume	2.6461×10^7

METHODOLOGY

- The burnup fuel is produced from the spent fuel of the Angra I reactor (PWR). The Angra I fuel has an initial enrichment of 3.1% which, after a burnup of 33 GWd/t, is then kept for another 5 years in a cooling pool. After that, the spent fuel matrix is reprocessed through the GANEX (Actinide Extraction Group) process and spiked with thorium [3].

➢ TRU-O_2 fuel

➢ TRU-N fuel

Table III: Fuel composition (normalized) TRU-O_2 with 15% of the fissile material [3].

Nuclide	Weight fraction	Nuclide	Weight fraction	Nuclide	Weight fraction
^{232}Th	6.62386E-01	^{239}Np	1.75072E-05	^{242}Cm	9.61704E-06
^{233}U	7.74502E-13	^{238}Pu	3.80787E-03	^{244}Cm	1.10250E-05
^{234}U	3.19641E-06	^{239}Pu	9.97122E-02	^{245}Cm	3.83639E-07
^{235}U	1.66354E-04	^{240}Pu	3.40877E-02	^{143}Nd	2.54401E-03
^{236}U	8.50376E-05	^{241}Pu	3.20511E-02	^{150}Sm	5.10164E-04
^{237}U	2.19361E-09	^{242}Pu	1.21128E-02	^{153}Eu	1.08484E-04
^{238}U	2.02225E-02	^{241}Am	3.08848E-05	^{16}O	1.20469E-01
^{237}Np	9.34742E-03	^{242}Am	5.68381E-08		
^{238}Np	2.77277E-07	^{243}Am	2.31640E-03		

Table IV: Fuel composition (normalized) TRU-N with 15% of the fissile material [3].

Nuclide	Weight fraction	Nuclide	Weight fraction	Nuclide	Weight fraction
^{232}Th	7.259827E-01	^{239}Np	1.752568E-05	^{242}Cm	9.627217E-06
^{233}U	7.753217E-13	^{238}Pu	3.811898E-03	^{244}Cm	1.103662E-05
^{234}U	3.199789E-06	^{239}Pu	9.981778E-02	^{245}Cm	3.840447E-07
^{235}U	1.665298E-04	^{240}Pu	3.412380E-02	^{143}Nd	2.546698E-03
^{236}U	8.512763E-05	^{241}Pu	3.208502E-02	^{150}Sm	5.107039E-04
^{237}U	2.195933E-09	^{242}Pu	1.212559E-02	^{153}Eu	1.085985E-04
^{238}U	2.024394E-02	^{241}Am	3.091746E-05	^{14}N	5.664246E-02
^{237}Np	9.357313E-03	^{242}Am	5.689827E-08		
^{238}Np	2.775700E-07	^{243}Am	2.318853E-03		

METHODOLOGY

- The simulations used the Monte Carlo N-Particle code (MCNP5), in which the parameters used for this analysis were 10^6 neutrons in 550 generations, with the first 50 discarded for source convergence.
  - Analyses the Beginning of Life (BOF) of the fuel;
  - Neutron flux in the volume of the transmutation layer.
- Burnup of fuel using Monteburns code, which links the MCNP to the depletion tool ORIGEN2.1
  - The burnup fuel process was conducted over 5 years, using a fission power of 1500 MW.

RESULTS AND DISCUSSION

Table V: Steady-state calculated system properties for TRU-O_2 and TRU-N fuels.

Parameters	TRU-O_2	TRU-N	Difference
k_{eff}	0.98793 ± 0.00069	0.96964 ± 0.00082	1829 pcm
$\bar{E}_{fission}$	4.3375E-01 MeV	4.5828E-01 MeV	5.35 %
η	1.5107	1.4598	3.37 %
Thermal neutron flux (0.625 eV)	0.00%	0.00%	0.0 pp
Intermediate neutron flux (0.625 eV - 100 keV)	45.61%	43.56%	2.05 pp
Fast neutron flux (>100 keV)	54.38%	56.44%	2.06 pp
P_{NL}	0.8262	0.8093	2.04 %
Percent of the rate elastic collision reaction	89.02 %	87.17%	1.85 pp

RESULTS AND DISCUSSION

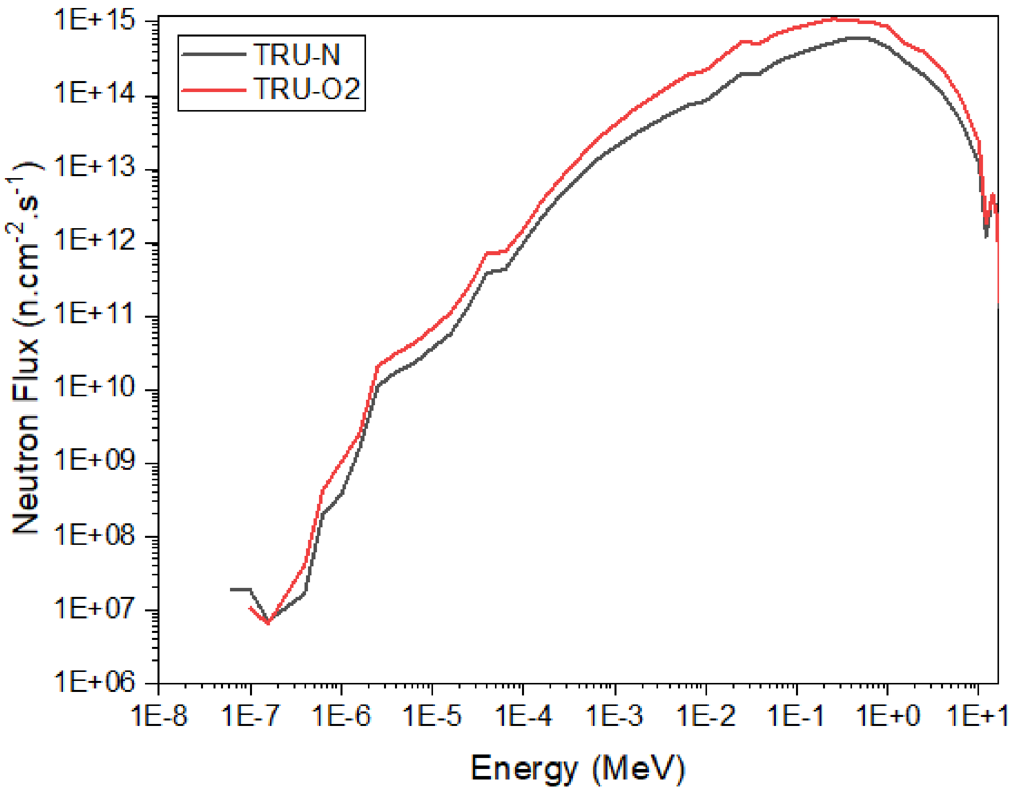

Figure 8 - Neutron flux calculated in the volume of the transmutation layer for TRU-N and TRU-O_2 models.

- Neutron flux in the volume of the transmutation layer.
- The neutron flux in TRU-O_2 is greater due to the higher k_{eff} value and the lower leakage in this system.

RESULTS AND DISCUSSION

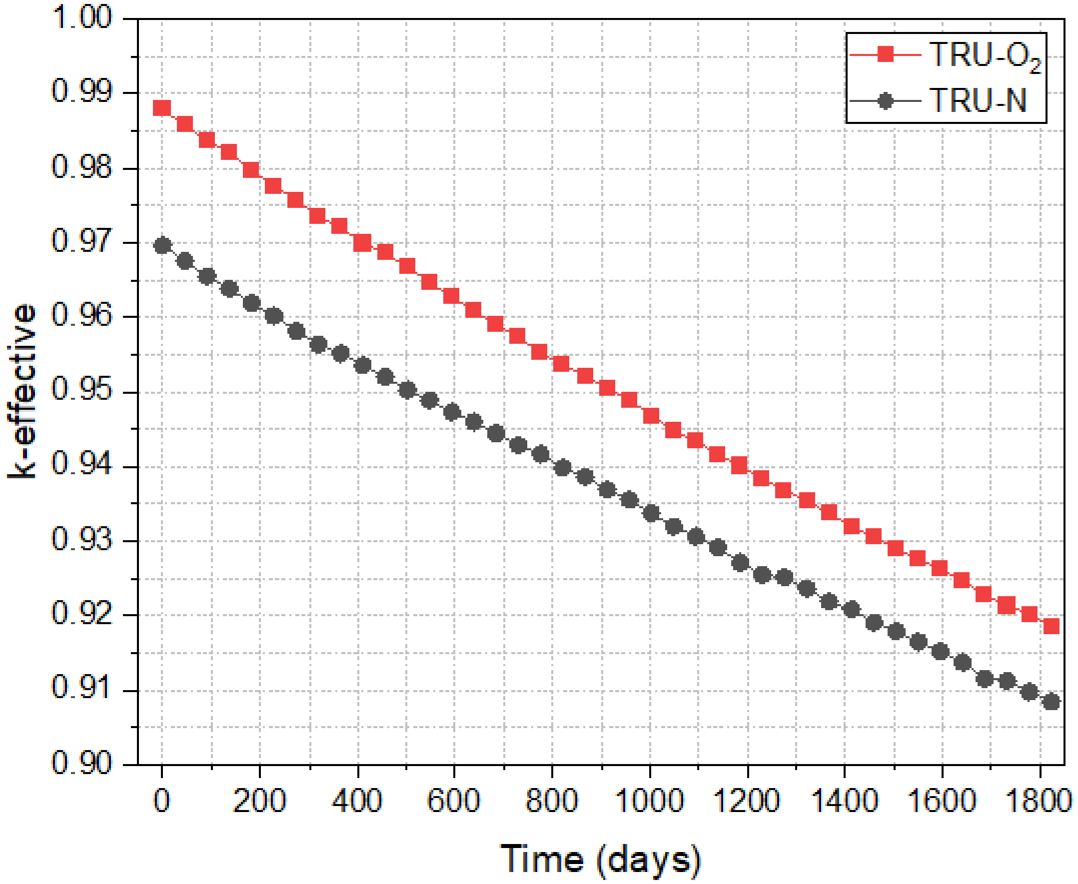

Figure 9 - k_{eff} values during fuel burnup for TRU-N and TRU-O_2 systems.

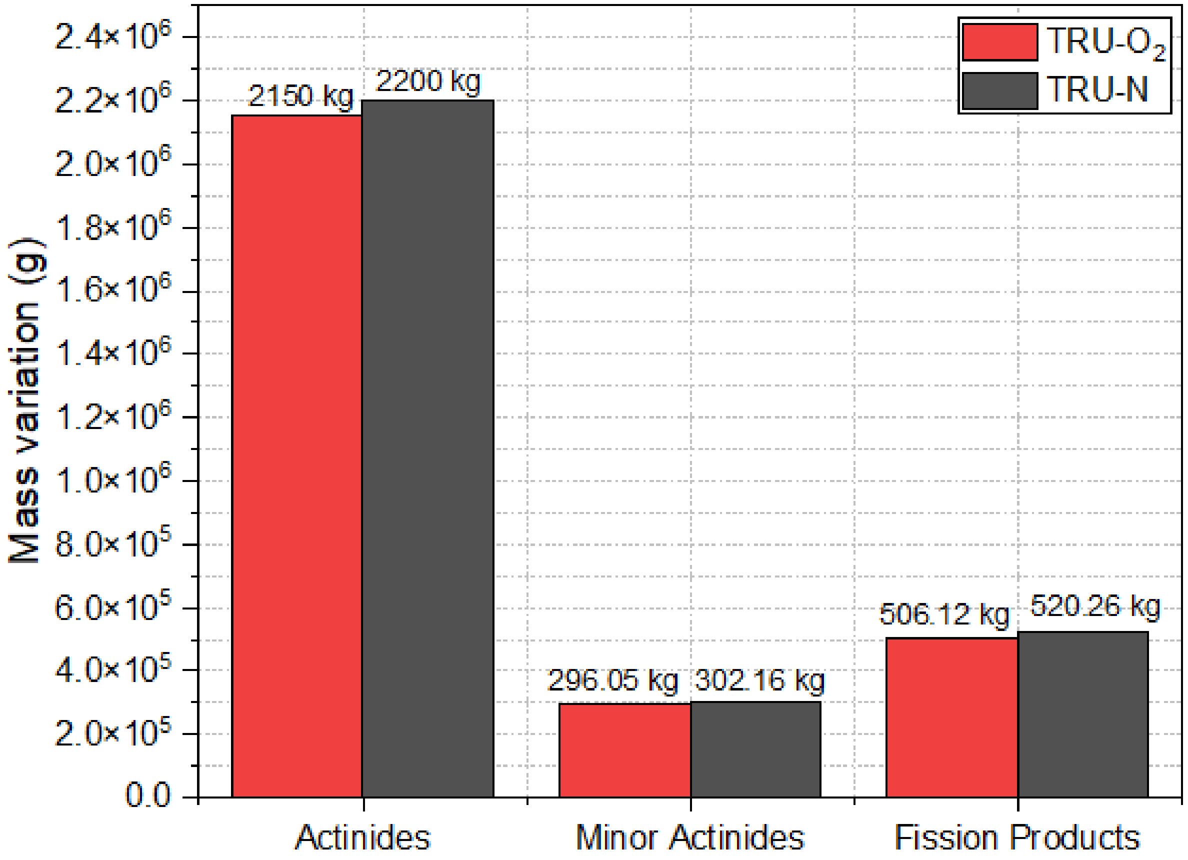

Figure 10 – Calculated transmutation of the actinides and minor actinides, and buildup fission products for TRU-N and TRU-O_2 systems.

CONCLUSIONS

- A previous analysis of TRU-N and TRU-O_2 fuels was carried out in a hybrid fusion-fission system based on the ARC reactor.

- TRU-O_2 and TRU-N fuels showed transmutation of actinides and minor actinides. TRU-N fuel presented optimal transmutation rates during burnup fuel, compared to TRU-O_2.

- Future work: Evaluate other fuels to be inserted into the hybrid fusion-fission system, (such as Uranium Carbide and Cermet fuel), and the most suitable coolant and cladding materials for these fuels.

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ACKNOWLEDGEMENTS

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