Monte Carlo simulations and experimental results on neutron production in the spallation target QUINTA irradiated with 660 MeV protons

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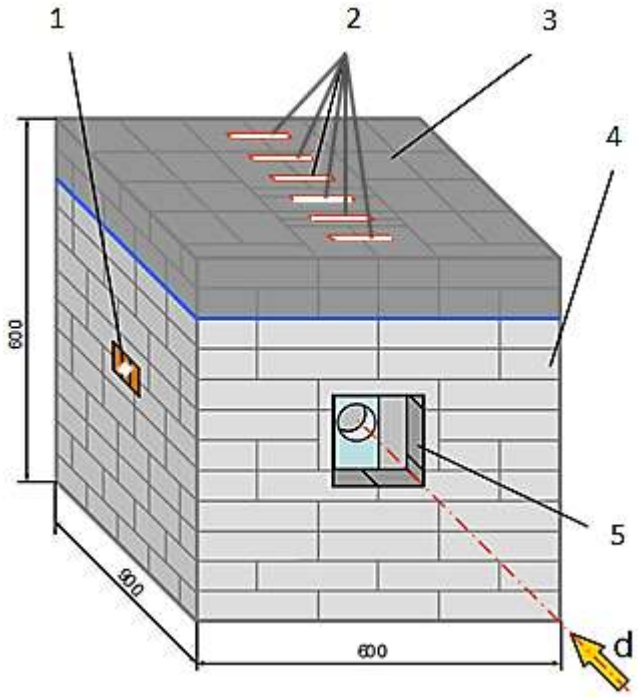
Highlights

The activation experiment was performed using the accelerated beam of Phasotron accelerator at the Joint Institute for Nuclear Research (JINR). The natural uranium spallation target QUINTA was irradiated with protons with energy 660 MeV. Monte Carlo simulations were performed using FLUKA and Geant4 codes. The number of leakage neutrons from the sections of the uranium target surrounded by the lead shielding and the number of leakage neutrons from the lead shield were determined. The total number of fissions in the setup QUINTA were determined. Experimental values of reaction rates for the produced nuclei in the ^{127}I sample were obtained and several values of reaction rates were compared with the results of simulations by FLUKA and Geant4 code. Experimentally determined fluence of neutrons in energy interval 10-200 MeV using the (n,xn) reactions in the ^{127}I(NaI) sample was compared with the results of simulations. Possibility of transmutation of the long-lived radionuclide ^{129}I in the QUINTA setup was estimated.

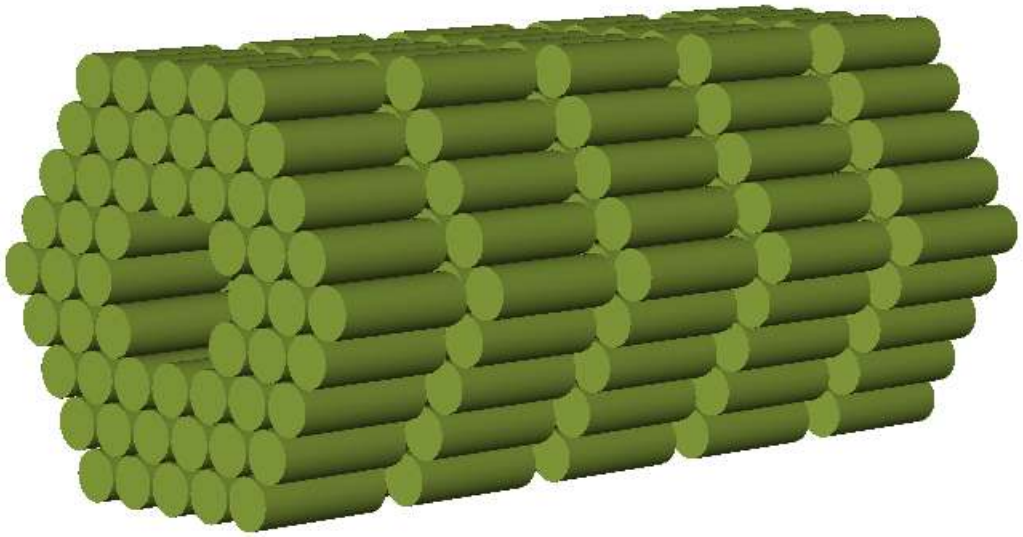
The general view of the QUINTA setup



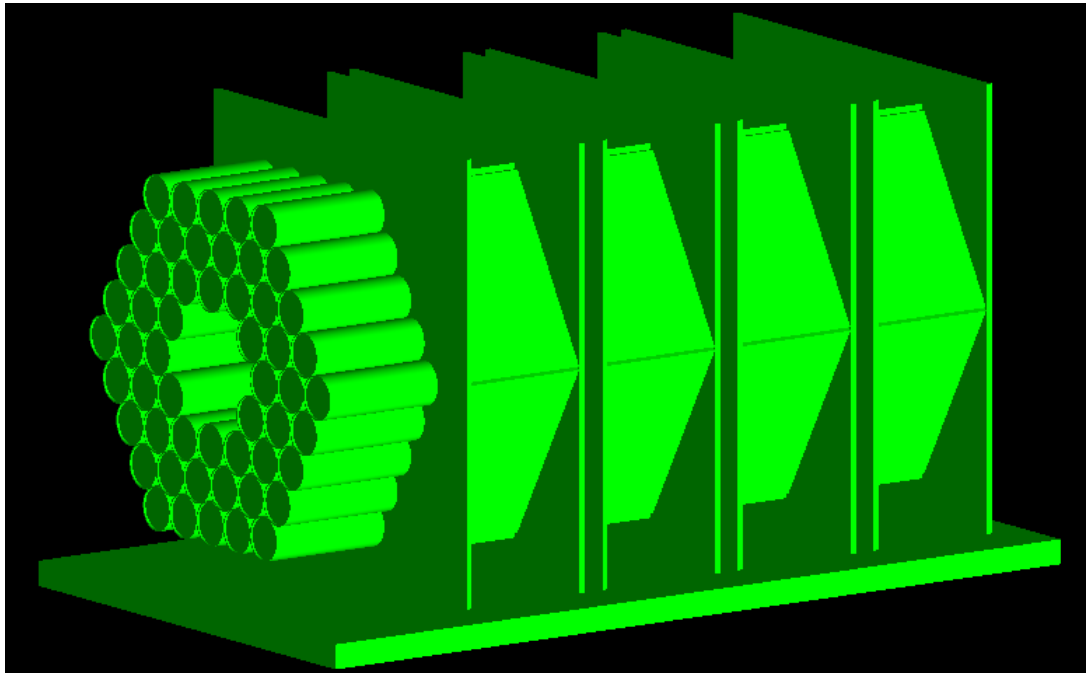
1 - window for placement of the transmutation samples, 2 - hatches for mounting and dismantling of detector probes and installation of the samples inside the uranium assemblies between sections, 3 - lead cover assembly, 4 - the lead assembly, 5 - the 15x15-cm^2 beam input window.

Protons energy, MeV	660	
Irradiation time, min.	300	
Integral number of protons	2.4(3)E+15	
Samples	I-127	I-129
Mass (I-129), g.	-	0.339
Mass (I-127), g.	1.270	0.074
Mass (Na-23), g.	0.230	0.067
Mass (Al-27), g.	-	17.6
Diameter of samples, cm	2.0	2.1

Part of models for the simulations by FLUKA code



Part of models for the simulations by Geant4 code



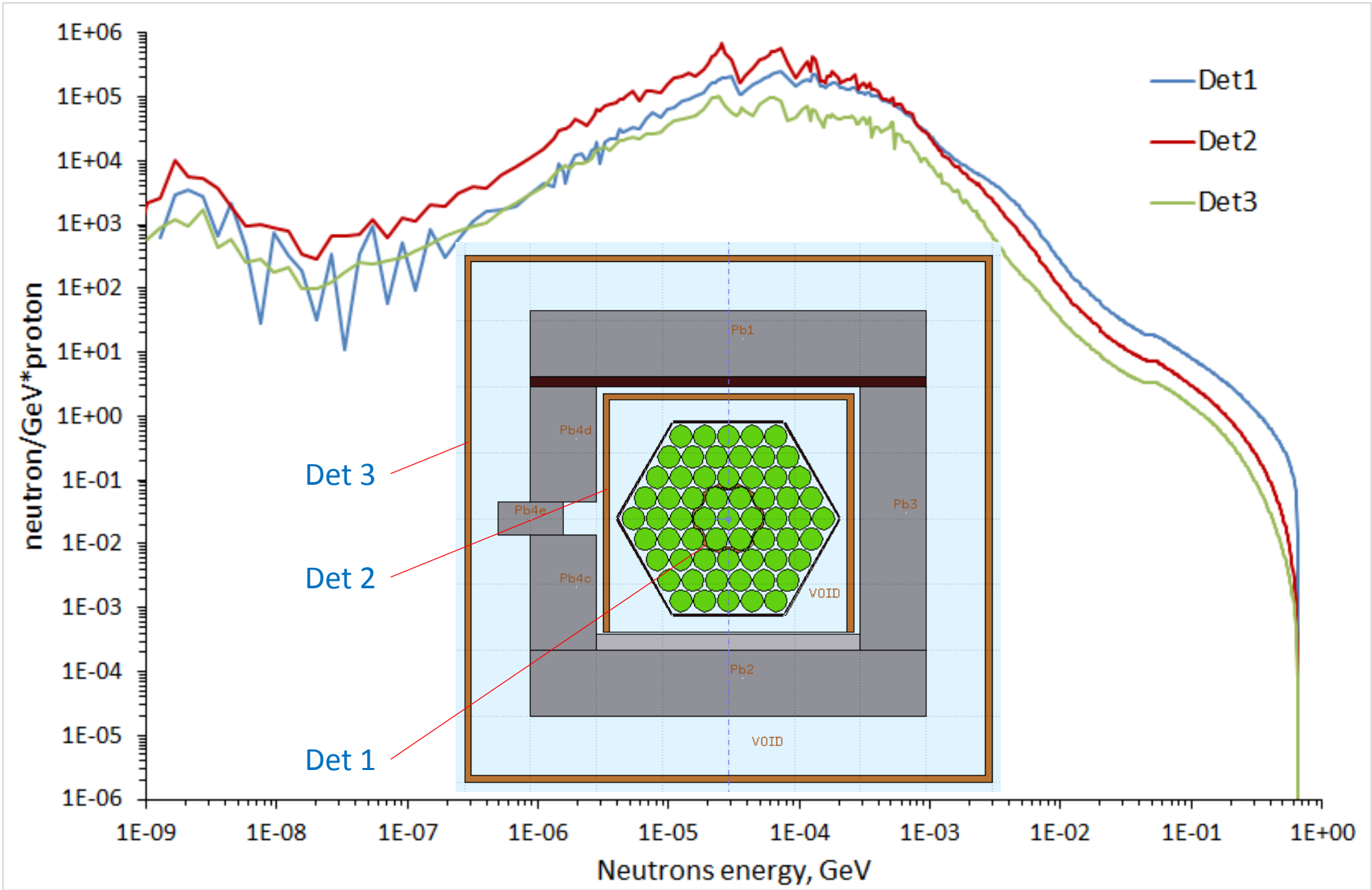
The Physics Models of FLUKA

The FLUKA hadron-nucleon interaction models are based on resonance production and decay at energies below a few GeV, and on the Dual Parton model at energies above.
Two models are used also in hadron-nucleus interactions.
At momenta below 3–5 GeV/c the PEANUT package includes a very detailed Generalised Intra-Nuclear Cascade (GINC) and a preequilibrium stage, while at high energies the Gribov-Glauber multiple collision mechanism is included in a less refined GINC.
Both modules are followed by equilibrium processes: evaporation, fission, Fermi break-up, γ-ray de-excitation.
Transport of neutrons with energies lower than a certain energy is performed in FLUKA by a multigroup algorithm.
The energy boundary below which multigroup transport takes over depends in principle on the cross section library used.
In the FLUKA neutron cross section library, the energy range up to 20 MeV is divided into 260 energy groups of approximately equal logarithmic width (31 of which are thermal).

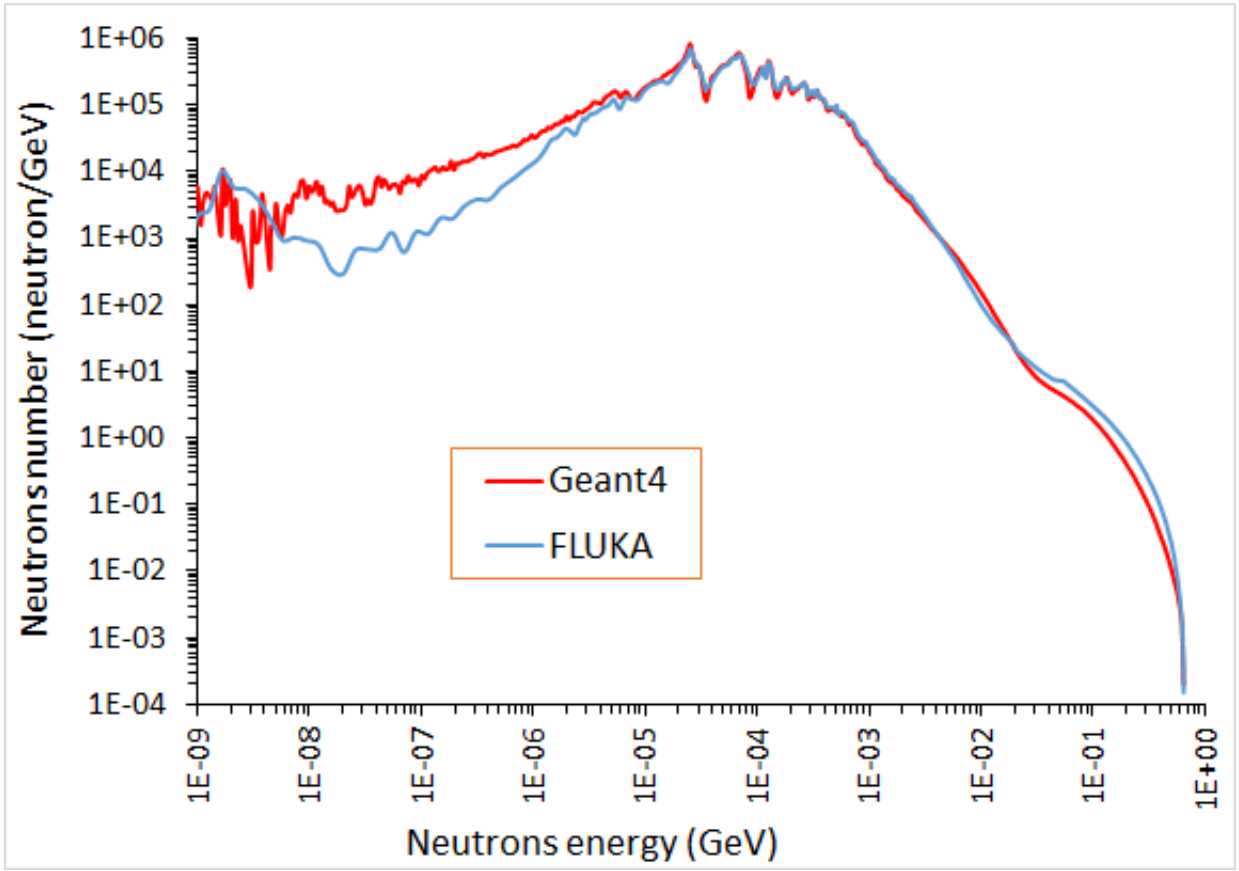
The Physics Models of Geant4

Bertini Intranuclear Cascade Model in Geant4 is a re-engineered version of the INUCL code and includes the Bertini intra-nuclear cascade model with excitons, a pre-equilibrium model, a nucleus explosion model, a fission model, and an evaporation model.
It treats nuclear reactions initiated by long-lived hadrons (p, n, π, K, Λ, Σ, Ξ, Ω) and γs with energies between 0 and 10 GeV.
In the simulations by the Geant4 code, the neutron transport class library simulates the interactions of neutrons with kinetic energies from thermal energies up to 20 MeV.
The interactions of neutrons at low energies are split into four parts in analogy to the other hadronic processes in Geant4. Radiative capture, elastic scattering, fission, and inelastic scattering are considered as separate models.
All cross-section data are taken from the ENDF/B-VI evaluated data library.
The high precision (HP) neutron models depend on the evaluated neutron data library (G4NDL) of cross sections, angular distributions and final state information and used when data are found in the library.
When data are missing, low energy parameterized neutron models are used and these alternative models cover the same types of interaction as the originals, that is elastic and inelastic scattering, capture and fission.

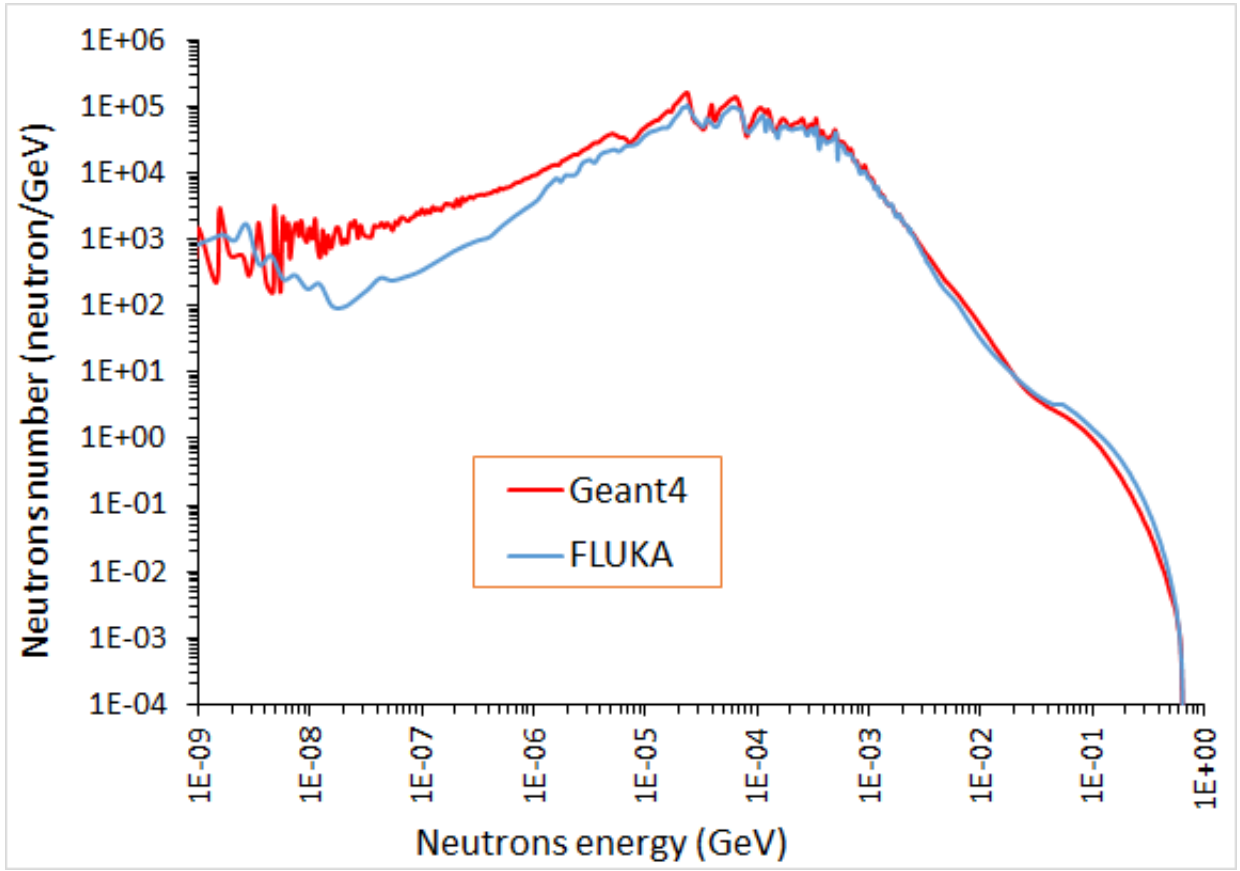
Results of simulations by FLUKA code for define fluence of leakage neutrons



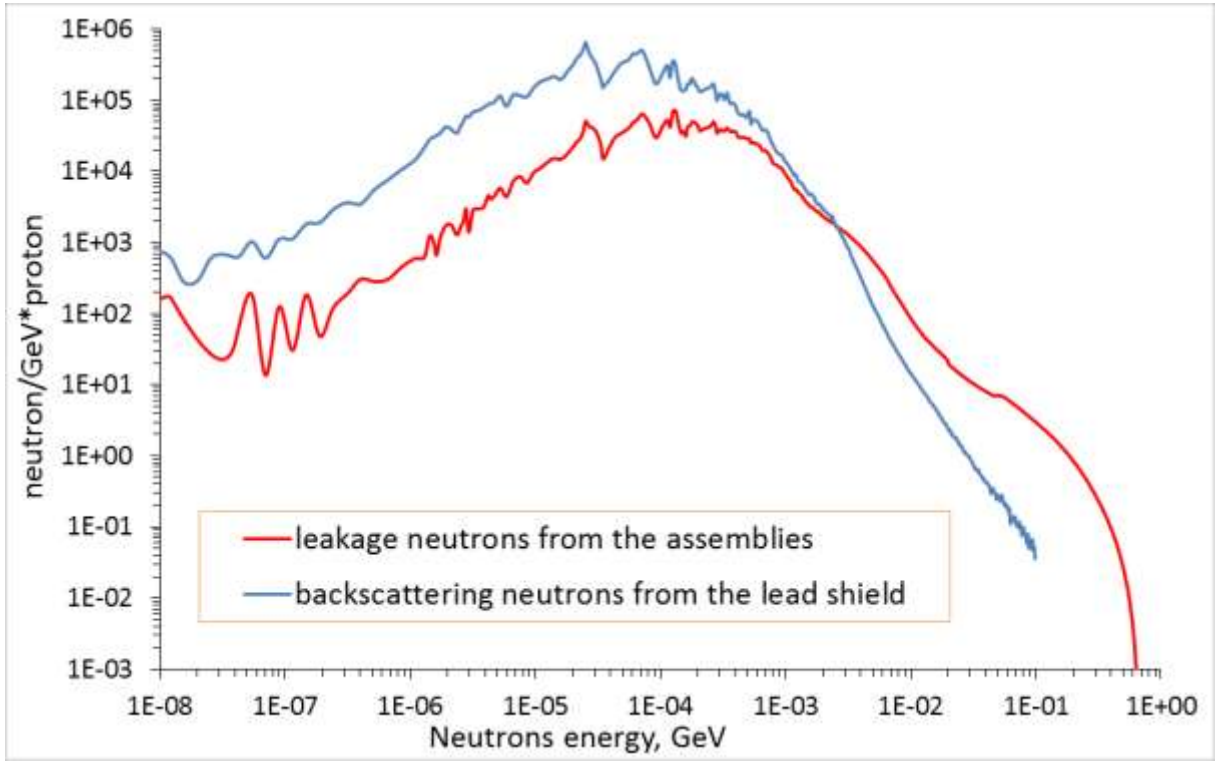
Summary number of leakage neutrons from the natural uranium assemblies and backscattering neutrons from the lead shield (Det2)



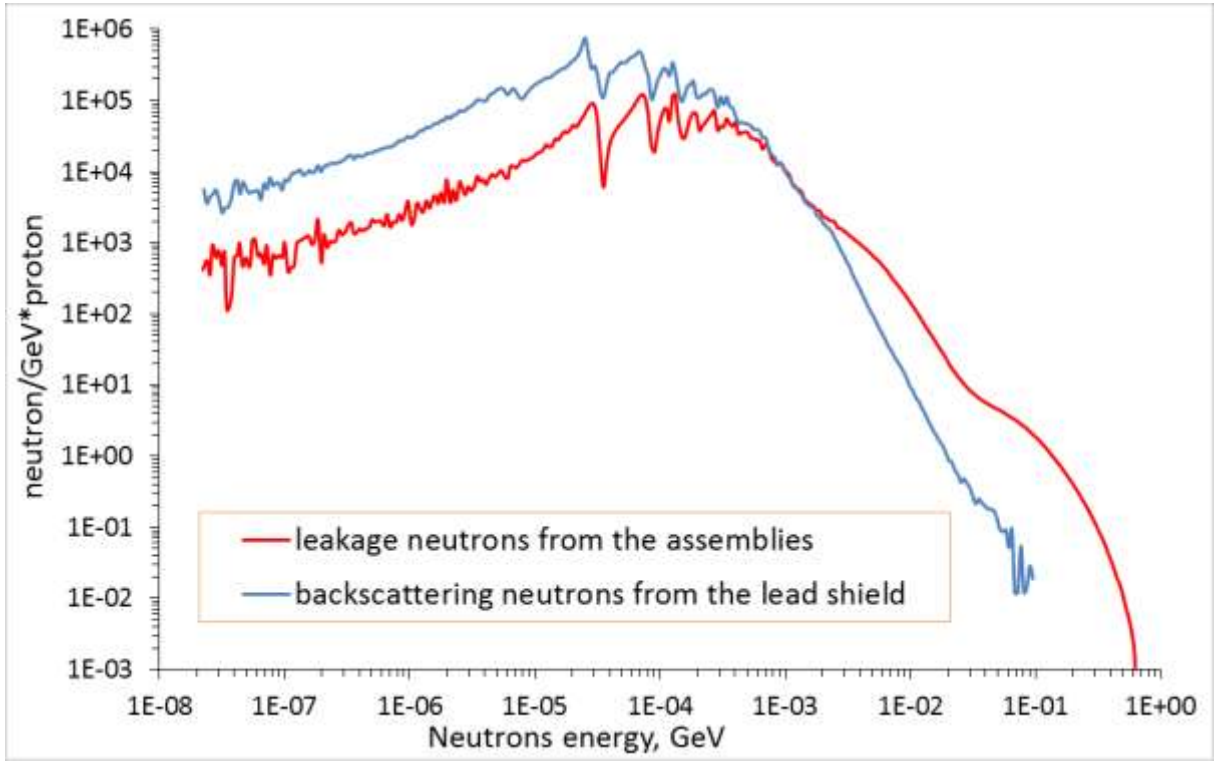
Number of leakage neutrons from the lead shield (Det3)



Number of leakage from the assemblies and backscattering neutrons from the lead shield (FLUKA simulations)



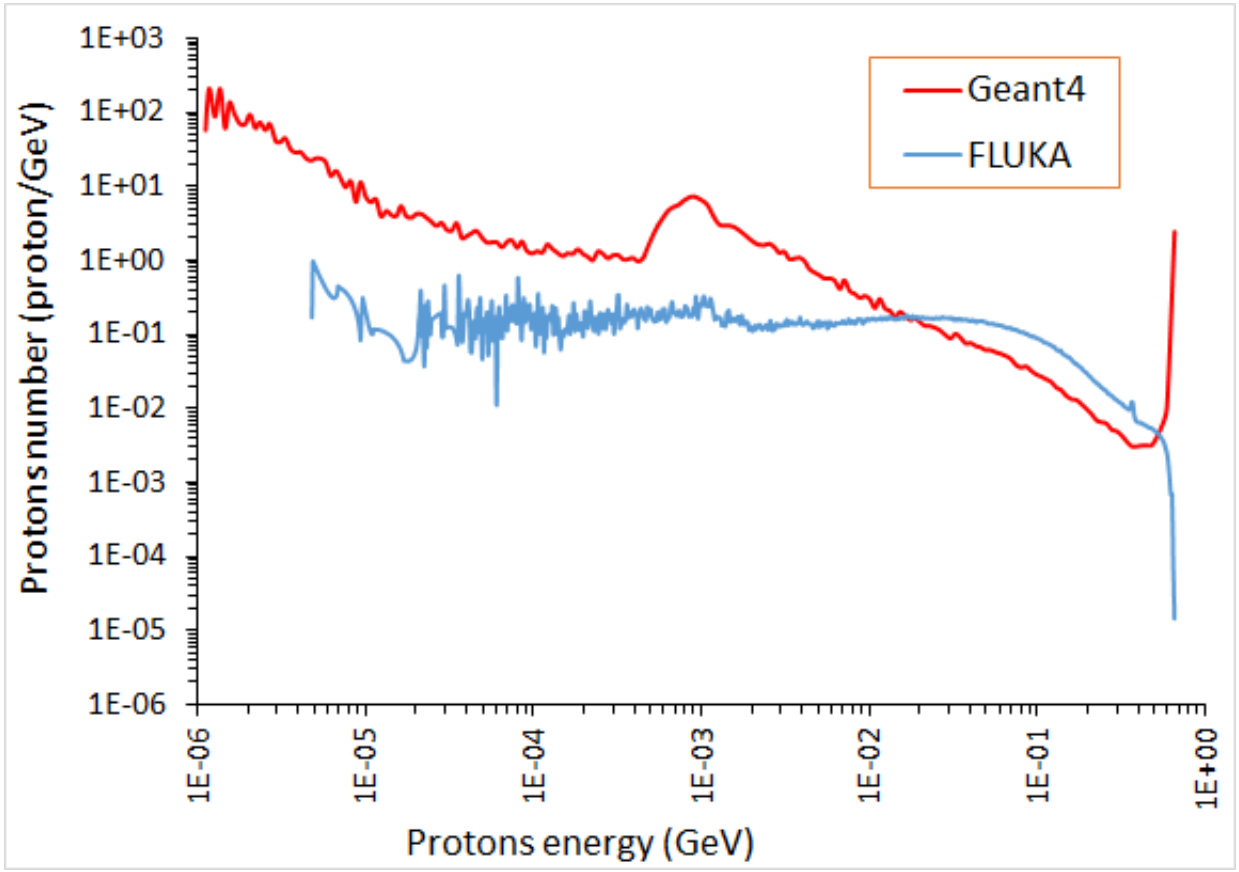
Number of leakage from the assemblies and backscattering neutrons from the lead shield (Geant4 simulations)



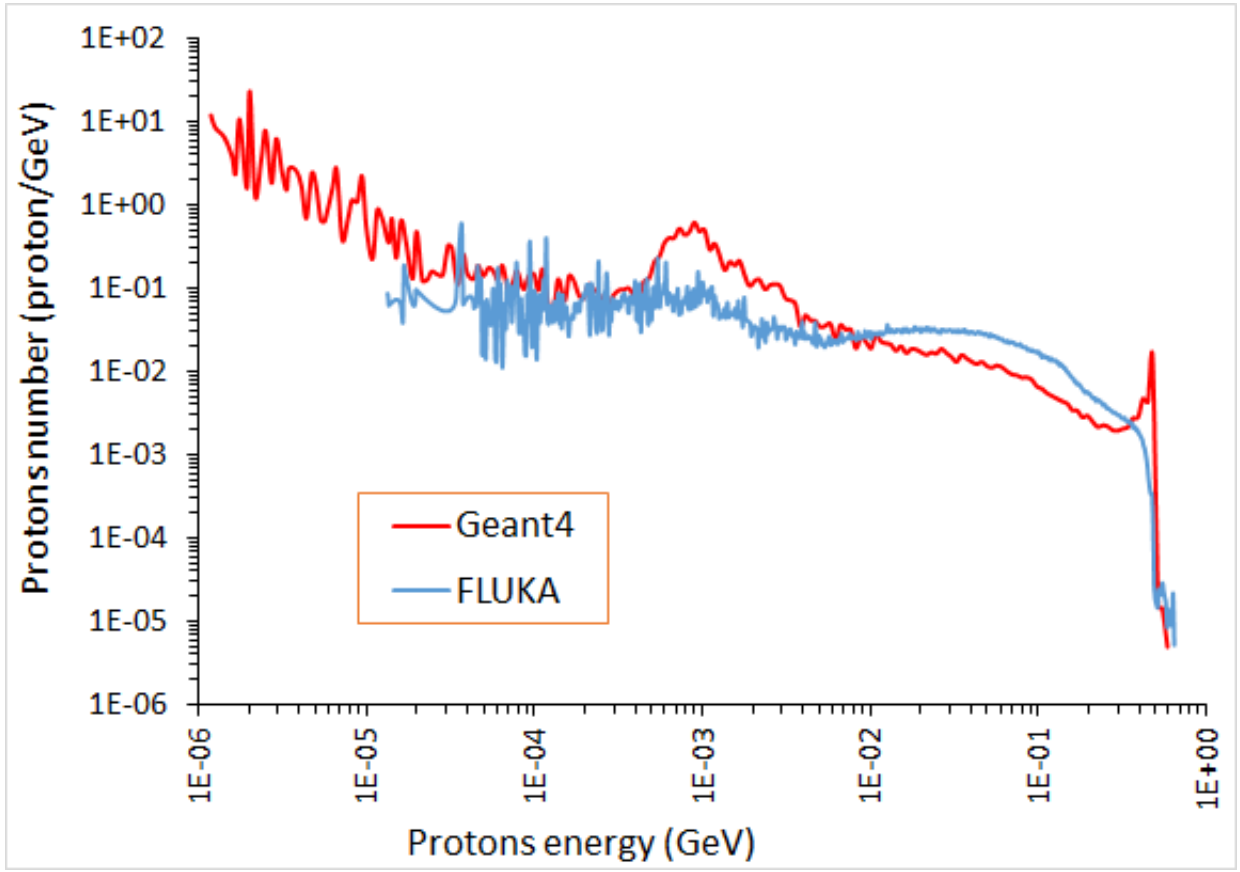
Results of simulations by FLUKA and Geant4 codes for define number of leakage neutrons

	Det 1		Det 2		Det 3	
	FLUKA	Geant4	FLUKA	Geant4	FLUKA	Geant4
Total number of neutrons	120	-	144	134	38	44
Number of neutrons with energy E > 1 MeV	31	-	19	17	7	7
Number of leakage neutrons from the assemblies			38	45		
Number of backscattering neutrons from the lead shield			106	89		

Number of leakage protons from the sections of natural uranium assemblies (Det2) in the lead shield



Number of leakage protons from the lead shield (Det3)



Definition number of fissions in the setup QUINTA

Total number of fissions in the setup per beam particle

5,52

Number of fissions with high energy particles (E > 20 MeV) in the setup per beam particle

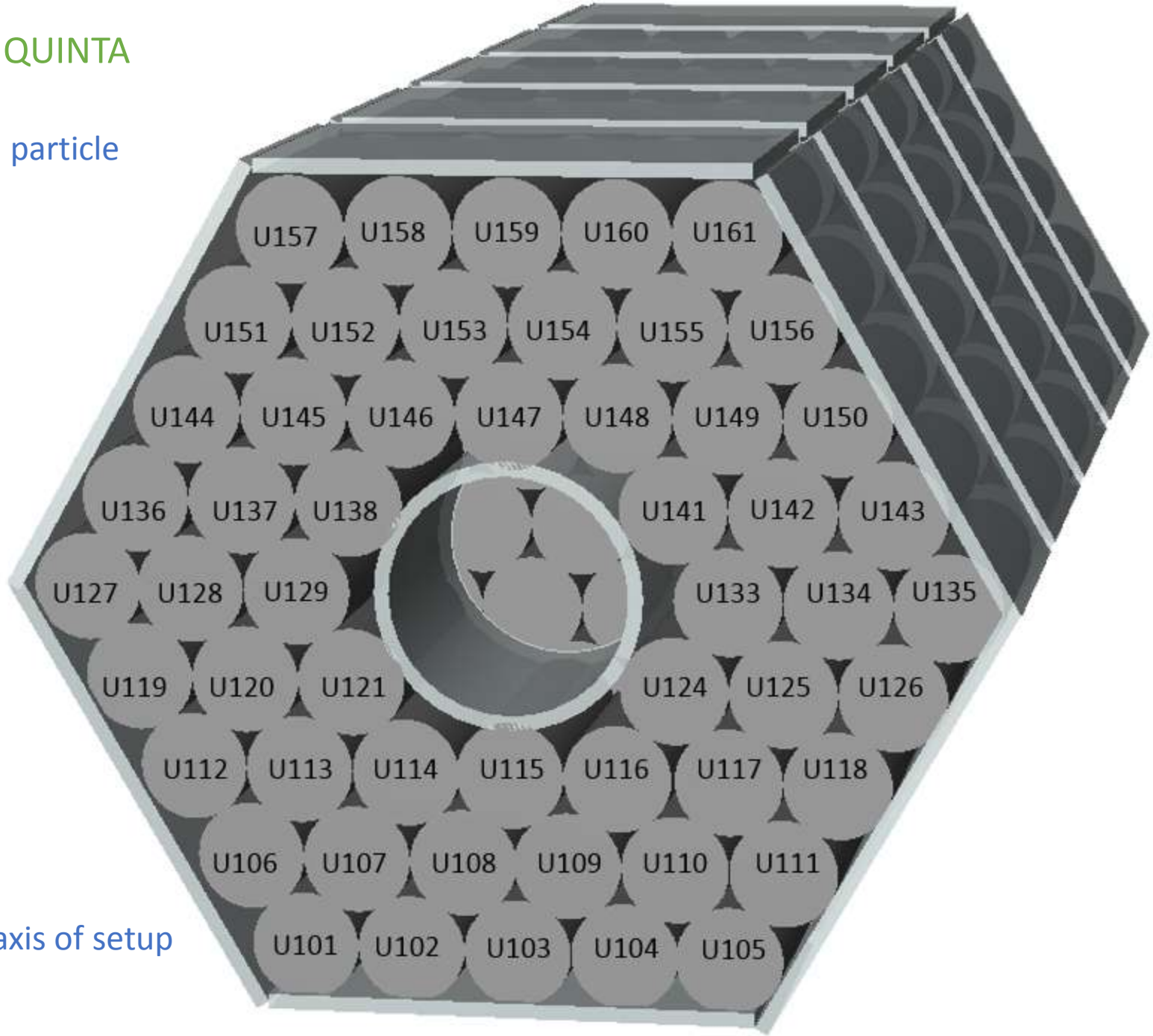
1,65 (100%)
0,79 (48,2%) generated by protons
0,84 (50,9%) generated by neutrons
5,59E-03 (0,3%) generated by pions+
1,08E-02 (0,7%) generated by pions-

Beam parameters for simulation

E(p) = 660 MeV
X(center) = 4,5 cm
Y(center) = 0 cm
X(FWHM) = 2,46 cm
Y(FWHM) = 3,19 cm

Angle between the beam axis and the central axis of setup

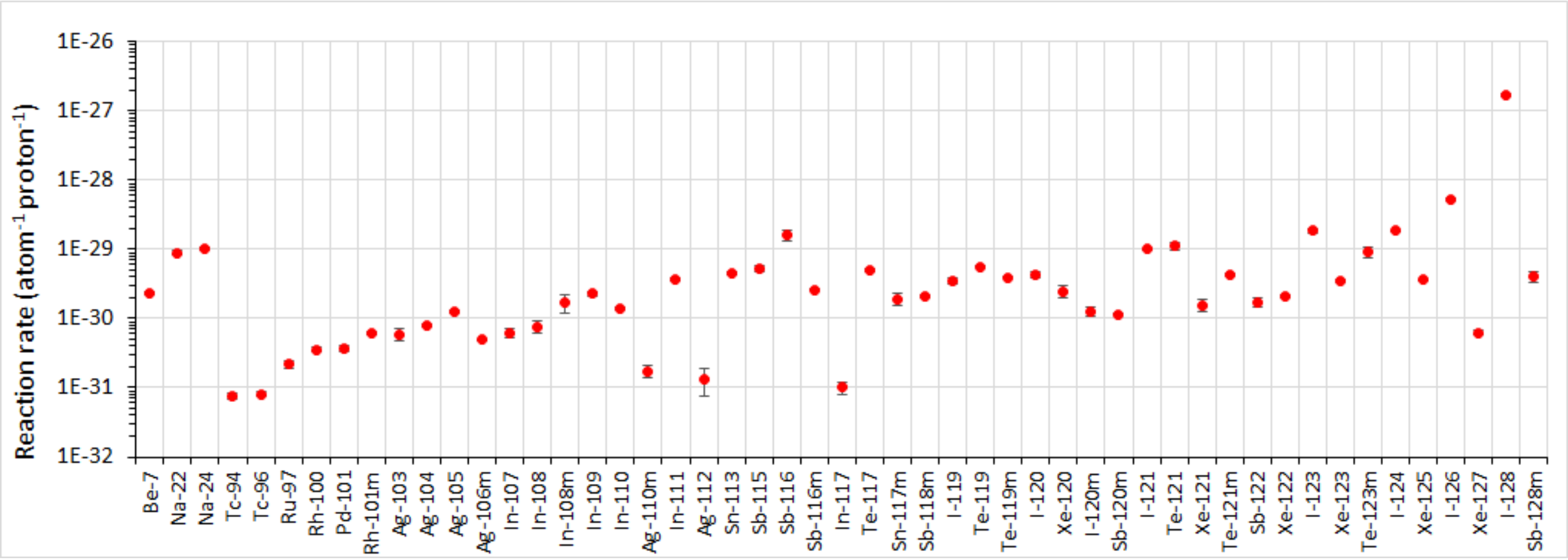
4 degree



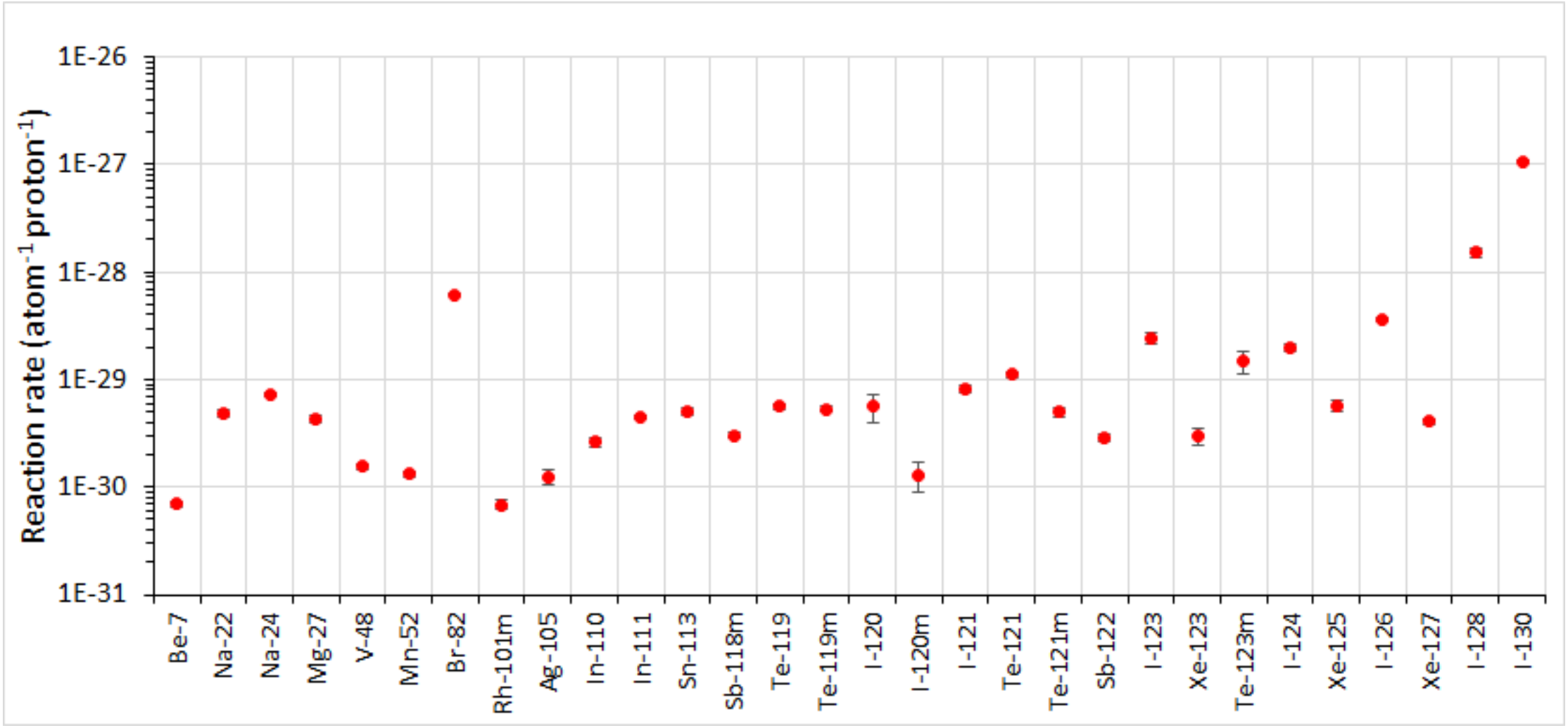
№ of U cylinders	№1 section	№2 section	№3 section	№4 section	№5 section
U01	4,15E-03	6,88E-03	7,07E-03	4,11E-03	1,86E-03
U02	5,25E-03	9,18E-03	9,31E-03	5,01E-03	1,96E-03
U03	6,09E-03	1,12E-02	1,06E-02	5,32E-03	2,14E-03
U04	6,46E-03	1,13E-02	1,02E-02	5,07E-03	2,03E-03
U05	5,83E-03	9,60E-03	8,45E-03	4,33E-03	1,88E-03
U06	4,84E-03	8,07E-03	8,43E-03	4,81E-03	2,03E-03
U07	6,97E-03	1,26E-02	1,25E-02	6,33E-03	2,47E-03
U08	8,81E-03	1,81E-02	1,64E-02	7,48E-03	2,61E-03
U09	1,03E-02	2,16E-02	1,78E-02	7,52E-03	2,67E-03
U10	1,03E-02	1,93E-02	1,54E-02	6,50E-03	2,38E-03
U11	7,99E-03	1,31E-02	1,09E-02	5,05E-03	2,02E-03
U12	4,87E-03	8,42E-03	8,80E-03	5,04E-03	2,04E-03
U13	8,04E-03	1,46E-02	1,47E-02	7,29E-03	2,54E-03
U14	1,26E-02	2,57E-02	2,38E-02	1,00E-02	3,12E-03
U15	1,70E-02	4,13E-02	3,25E-02	1,14E-02	3,36E-03
U16	2,00E-02	4,65E-02	2,99E-02	1,01E-02	3,18E-03
U17	1,61E-02	2,94E-02	2,00E-02	7,53E-03	2,58E-03
U18	9,78E-03	1,64E-02	1,24E-02	5,45E-03	2,12E-03
U19	4,62E-03	7,67E-03	7,95E-03	4,64E-03	2,03E-03
U20	7,67E-03	1,37E-02	1,40E-02	7,10E-03	2,59E-03
U21	1,36E-02	2,73E-02	2,61E-02	1,14E-02	3,17E-03
U22		6,34E-02	6,00E-02	2,06E-02	4,45E-03
U23		1,86E-01	8,71E-02	1,76E-02	4,09E-03
U24	9,11E-02	9,13E-02	4,12E-02	1,13E-02	3,18E-03
U25	2,09E-02	3,45E-02	2,12E-02	7,83E-03	2,46E-03
U26	9,91E-03	1,57E-02	1,19E-02	5,16E-03	2,05E-03
U27	3,82E-03	6,04E-03	6,33E-03	4,00E-03	1,76E-03
U28	6,32E-03	1,07E-02	1,12E-02	5,96E-03	2,34E-03
U29	1,18E-02	2,07E-02	2,06E-02	9,42E-03	2,98E-03
U30		4,75E-02	4,76E-02	1,71E-02	3,78E-03
U31		2,60E-01	1,95E-01	2,64E-02	4,60E-03
U32		6,08E-01	9,88E-02	1,62E-02	3,96E-03
U33	8,03E-02	7,39E-02	3,34E-02	9,98E-03	2,91E-03

№ of U cylinders	№1 section	№2 section	№3 section	№4 section	№5 section
U34	1,62E-02	2,56E-02	1,69E-02	6,57E-03	2,31E-03
U35	7,87E-03	1,15E-02	9,41E-03	4,38E-03	1,83E-03
U36	4,63E-03	7,42E-03	8,09E-03	4,74E-03	2,00E-03
U37	7,62E-03	1,40E-02	1,40E-02	7,02E-03	2,58E-03
U38	1,34E-02	2,72E-02	2,60E-02	1,14E-02	3,20E-03
U39		6,33E-02	5,98E-02	2,07E-02	4,52E-03
U40		1,85E-01	8,65E-02	1,77E-02	3,96E-03
U41	9,06E-02	9,15E-02	4,11E-02	1,15E-02	3,20E-03
U42	2,09E-02	3,45E-02	2,12E-02	7,87E-03	2,55E-03
U43	9,87E-03	1,53E-02	1,17E-02	5,01E-03	2,09E-03
U44	4,97E-03	8,52E-03	8,68E-03	4,95E-03	2,07E-03
U45	8,16E-03	1,48E-02	1,44E-02	7,33E-03	2,63E-03
U46	1,26E-02	2,60E-02	2,34E-02	1,01E-02	3,11E-03
U47	1,68E-02	4,16E-02	3,28E-02	1,16E-02	3,34E-03
U48	1,96E-02	4,61E-02	2,98E-02	1,01E-02	3,10E-03
U49	1,61E-02	2,93E-02	1,99E-02	7,59E-03	2,51E-03
U50	1,00E-02	1,61E-02	1,23E-02	5,38E-03	2,07E-03
U51	4,81E-03	8,10E-03	8,18E-03	4,69E-03	1,99E-03
U52	6,87E-03	1,27E-02	1,24E-02	6,19E-03	2,36E-03
U53	8,71E-03	1,78E-02	1,61E-02	7,47E-03	2,56E-03
U54	1,03E-02	2,14E-02	1,77E-02	7,50E-03	2,55E-03
U55	1,02E-02	1,94E-02	1,54E-02	6,60E-03	2,37E-03
U56	7,85E-03	1,29E-02	1,09E-02	5,07E-03	1,98E-03
U57	4,07E-03	6,69E-03	6,99E-03	4,00E-03	1,81E-03
U58	5,13E-03	9,06E-03	9,02E-03	4,83E-03	2,05E-03
U59	6,13E-03	1,10E-02	1,03E-02	5,21E-03	2,05E-03
U60	6,30E-03	1,12E-02	1,02E-02	4,95E-03	2,01E-03
U61	5,82E-03	9,48E-03	8,43E-03	4,34E-03	1,85E-03
Fissions in the section	0,75	2,59	1,51	0,51	0,16
Total number of fissions in the setup per beam particle				5,52	
Number of high energy fissions (E(n,p) > 20 MeV) in the setup per beam particle				1,65	

Experimental cumulative values of reaction rate for produced nucleus in the ^{127}I sample



Experimental cumulative values of reaction rate for produced nucleus in the ^{129}I sample

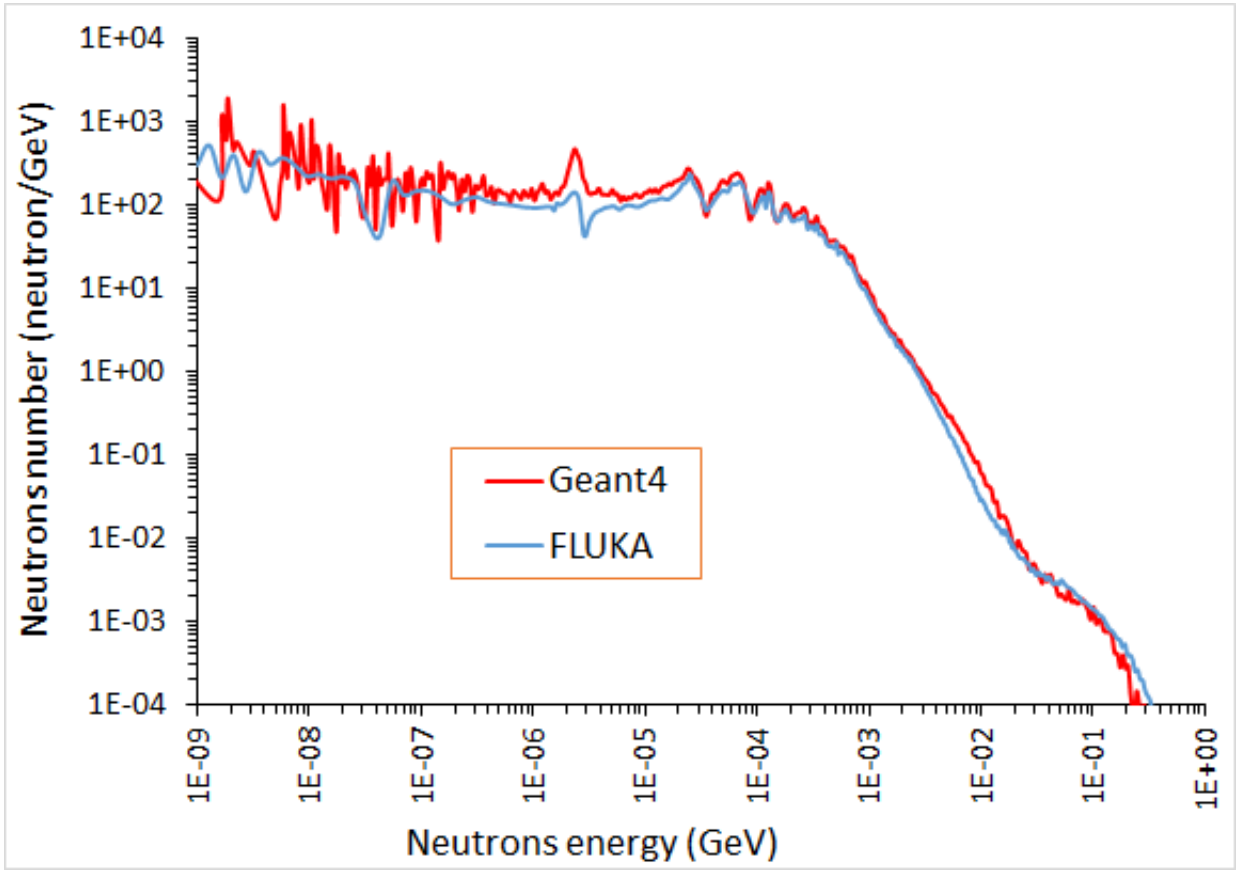


Independent values of reaction rates for the products of reactions $^{127}I(n,\gamma)$ and $^{127}I(n,xn)$ and comparison with the results of simulations

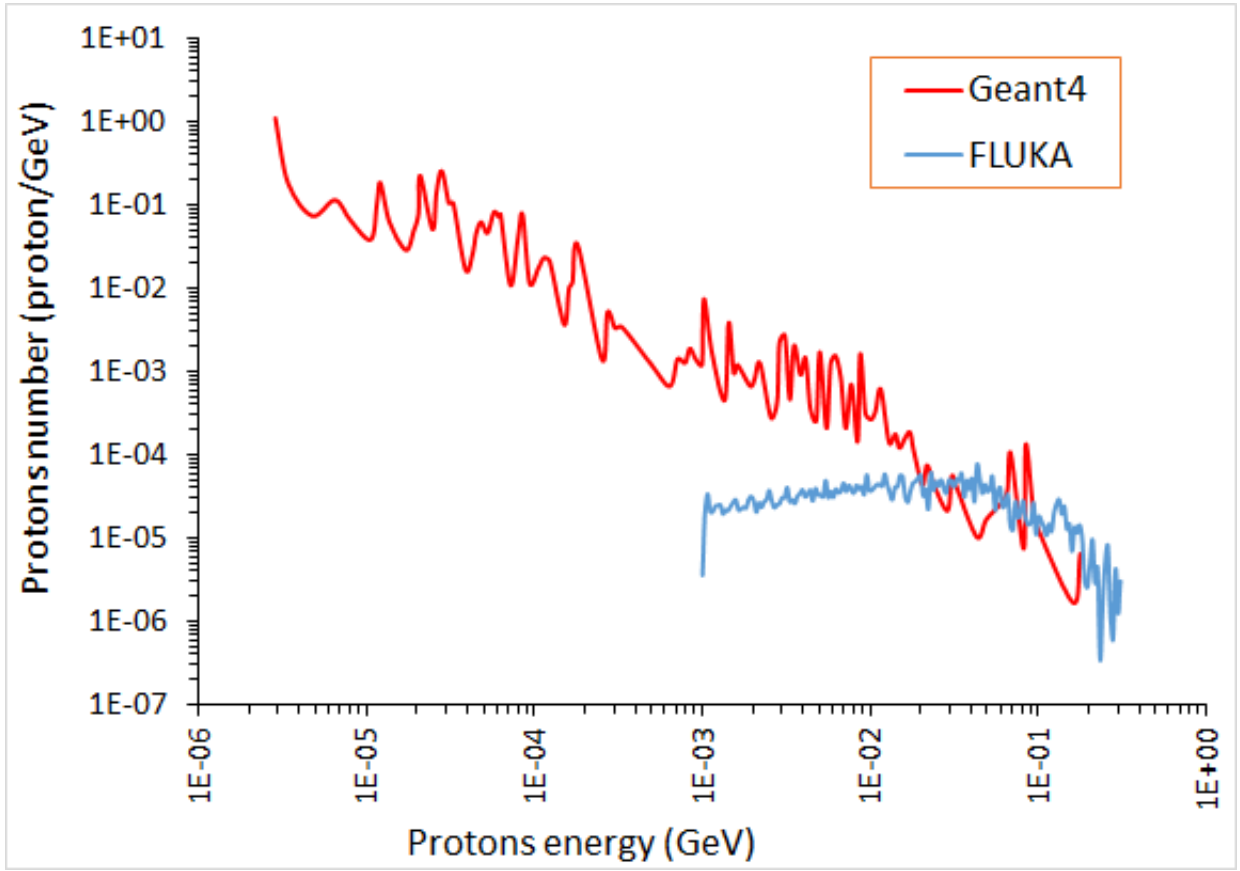
Nuclear reactions	Experimental (independent) reaction rate	Exp./Calc. (FLUKA)	Exp./Calc. (Geant4)
$^{127}I(n,\gamma)^{128}I$	1.68(9)E-27	0.36(1)	0.21(1)
$^{127}I(n,2n)^{126}I$	5.28(23)E-29	1.02(4)	0.41(2)
$^{127}I(n,4n)^{124}I$	1.86(4)E-29	1.14(2)	0.90(2)
$^{127}I(n,5n)^{123}I$	1.50(8)E-29	1.19(6)	0.91(5)
$^{127}I(n,7n)^{121}I$	8.43(36)E-30	2.71(12)	1.34(6)
$^{127}I(n,8n)^{120}I$	1.77(18)E-30	1.42(14)	0.69(7)
$^{127}I(n,9n)^{119}I$	1.47(14)E-30	1.89(18)	1.04(10)

We assume that for $^{127}I(n,\gamma)$ reaction, the difference between the results of simulations and experiment is caused by insufficient statistics for the accurate determination of the low energy (up to 1 keV) neutron fluence, which have a high capture cross section for ^{127}I nuclei.

Fluence of neutrons in the ^{127}I sample



Fluence of protons in the ^{127}I sample



Definition of neutron fluence using ^{127}I(n,xn) reactions

For experimentally definition of neutron fluence we are used of 80% from experimental independent values of reaction rate, because, results of simulations shows that neutrons in energy interval ΔE adding 80(3)% contribution to the reaction rate. Neutron fluence is defined by solving the equation (1) and averaged values of the cross sections for energy interval ΔE are obtained from the results of simulations by FLUKA code.

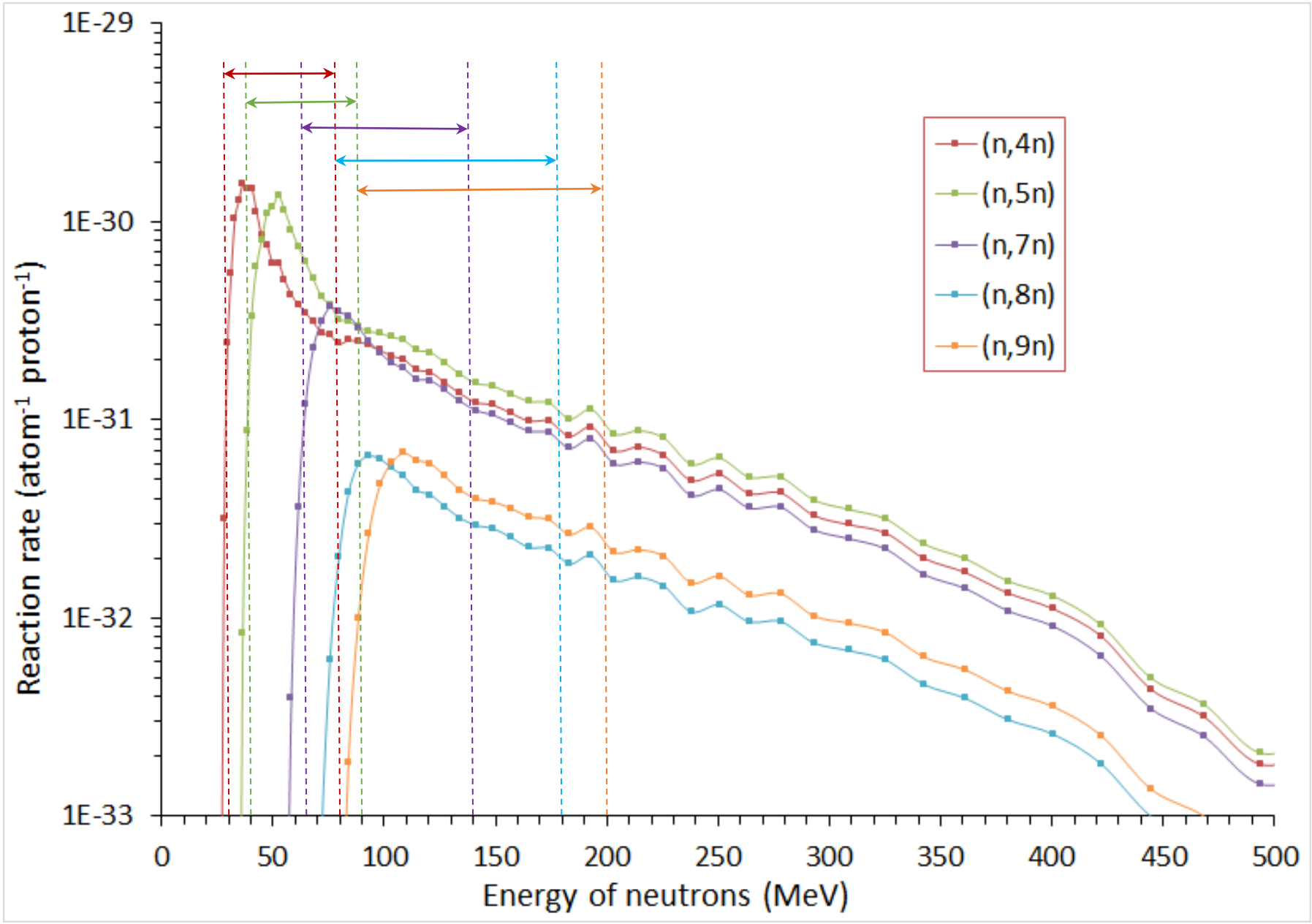
$$R(A_T, Z_T) = \int_{E_{thr}(A_T,Z_T)}^{E_{max}} \sigma(A_T, Z_T, E_n)\varphi(E_n)dE_n \qquad (1)$$

$\sigma(A_T, Z_T, E_n)$ is the reaction cross section, $\varphi(E_n)$ the neutron fluence.

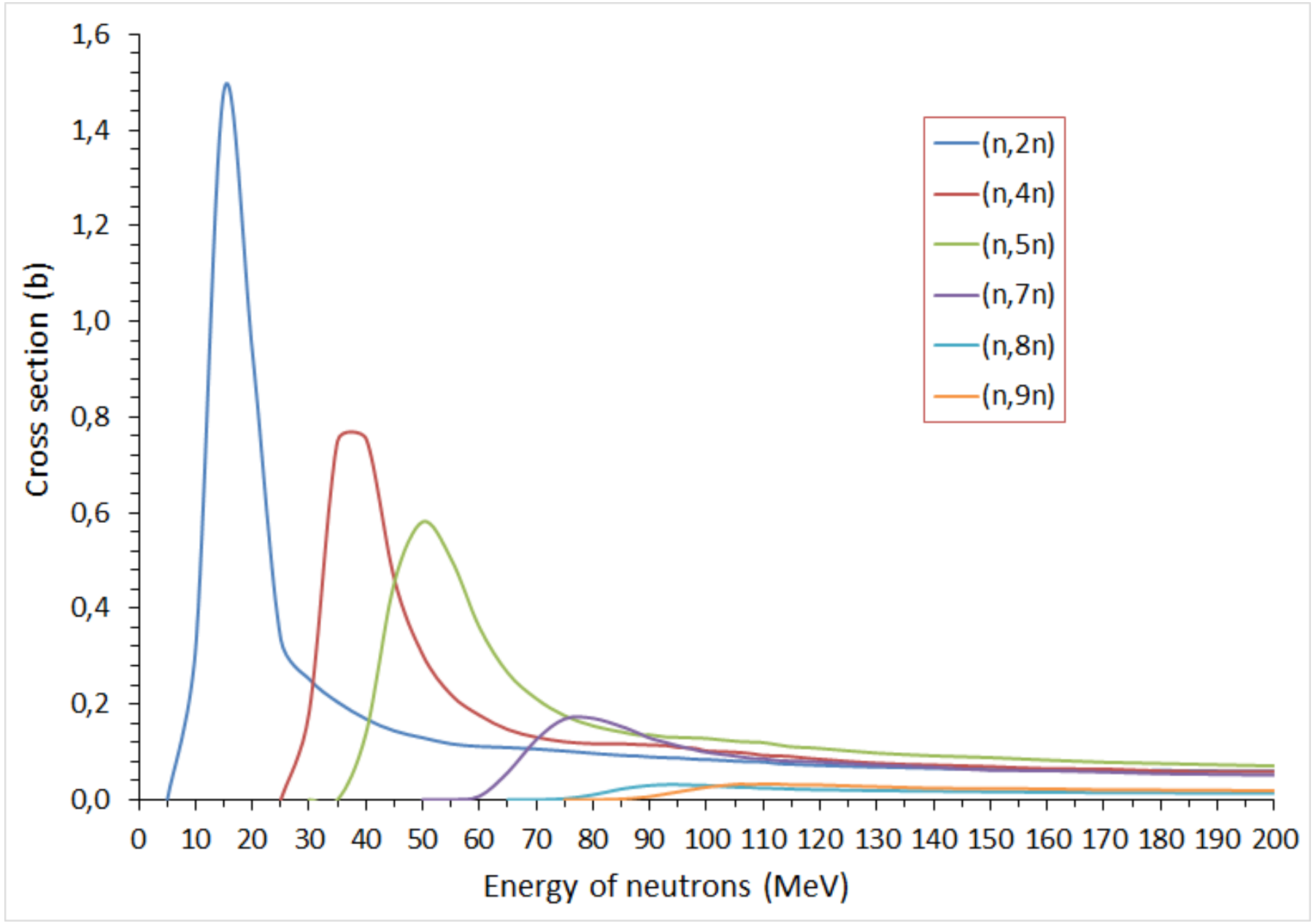
Nuclear reactions	0.8R	ΔE (MeV)	σ (b)	F
^{127}I(n,2n)^{126}I	4.23(19)E-29	10 – 30	0.668	3.16(63)E-6
^{127}I(n,4n)^{124}I	1.49(3)E-29	30 – 80	0.306	9.7(19)E-7
^{127}I(n,5n)^{123}I	1.20(7)E-29	40 – 90	0.282	8.5(17)E-7
^{127}I(n,7n)^{121}I	6.75(29)E-30	65 – 140	0.105	8.5(17)E-7
^{127}I(n,8n)^{120}I	1.42(14)E-30	80 – 180	0.021	6.8(14)E-7
^{127}I(n,9n)^{119}I	1.17(11)E-30	90 – 200	0.023	4.62(92)E-7

0.8R – 80% of independent values of reaction rate R(exp.), σ – averaged values of reaction cross sections for energy interval ΔE, F – fluence of neutrons (n/MeV·p·cm^2).

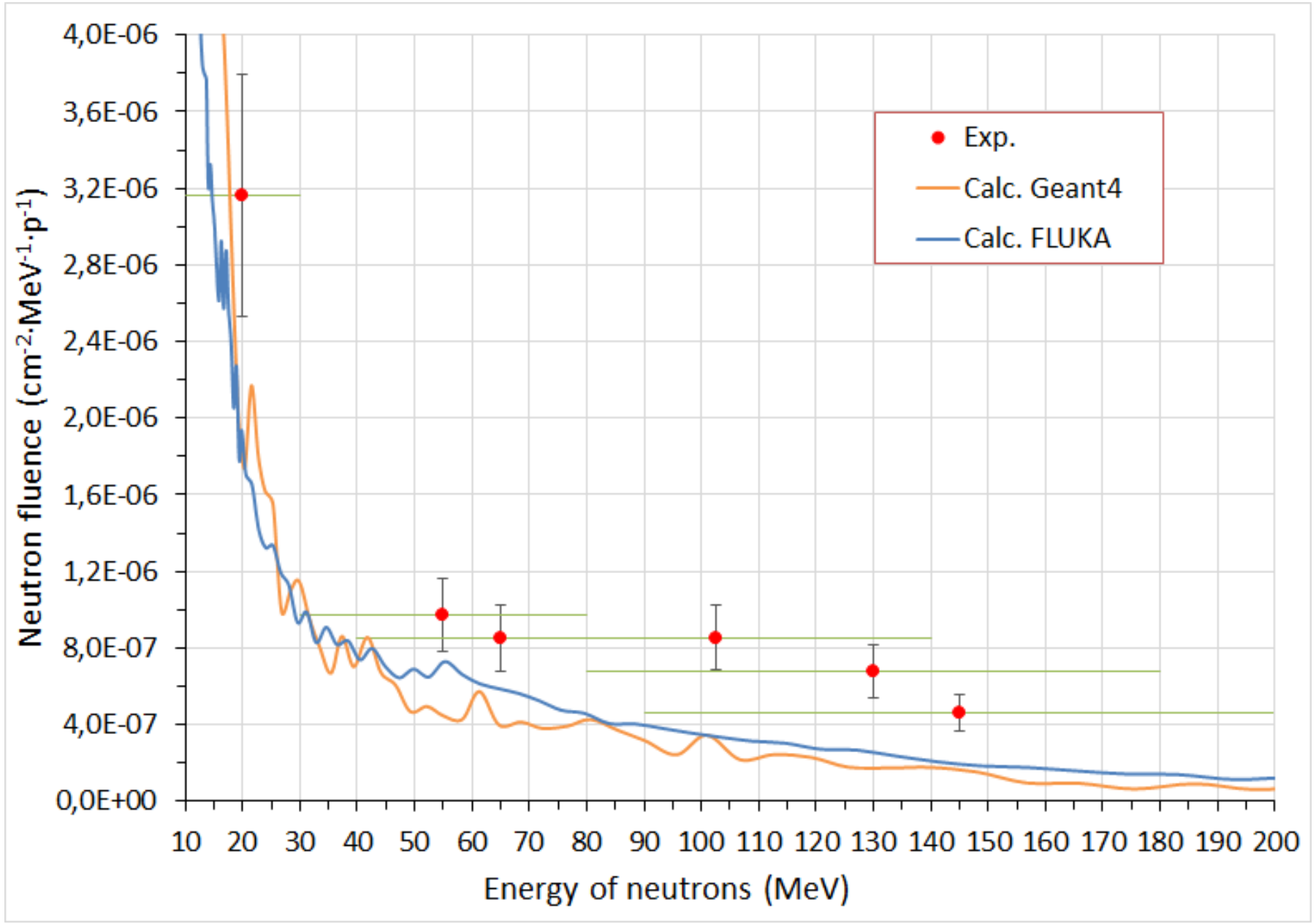
Dependence of ^{127}I(n,xn) reaction rates on energy of incident neutrons (simulations by FLUKA code)



Cross sections for ^{127}I(n,xn) reactions (simulations by FLUKA code)



Experimental and calculated values of neutron fluence in the ^{127}I sample



Conclusions

- The natural uranium spallation target QUINTA was irradiated with protons of energy 660 MeV.
- The simulations were performed by FLUKA and Geant4 codes.
- Results of simulations by FLUKA and Geant4 codes on the leakage of neutrons from the sections of uranium assembly and from the surface of the QUINTA setup are in good agreement.
- In the case of ^{127}I(NaI) sample, ratios of the experimental reaction rates to results of simulations by FLUKA and Geant4 codes for the (n,γ) reaction are 0.36 and 0.21 and for the (n,xn) reactions are in the interval 0.41-2.71.
- The obtained results for the determination of neutron fluence, using ^{127}I(n,xn) reactions in the experiment, have reasonable agreement with the neutron spectra obtained from simulations by FLUKA and Geant4 codes.
- Determined ratio of the mass of produced ^{130}I to mass of ^{129}I, in order to experimentally assess the possibility of transmutation by ^{129}I(n,γ) reaction. The ratio is 2.59(4)E-12 for the case of 5-hour irradiation time with 1.33E+11 proton/sec beam intensity.

Thank you for your attention