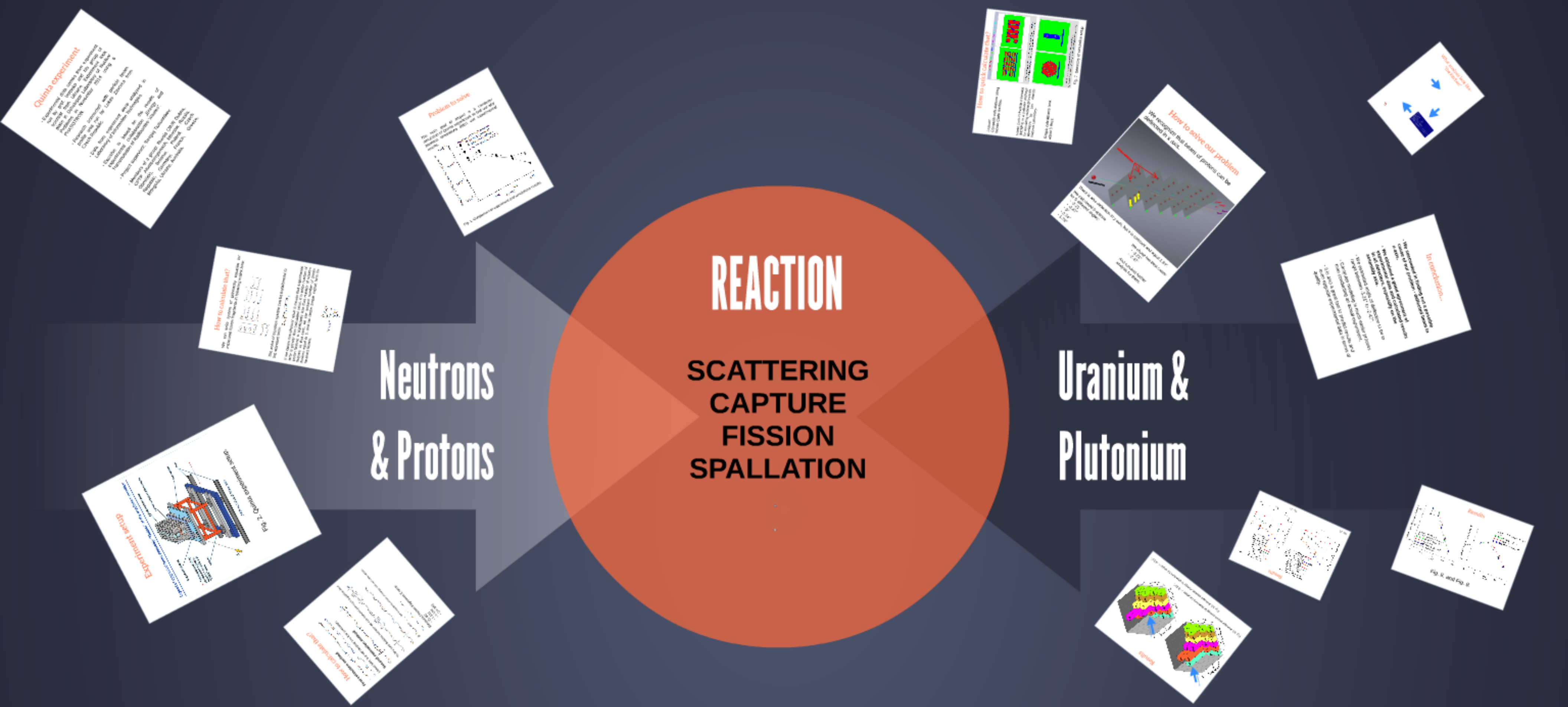
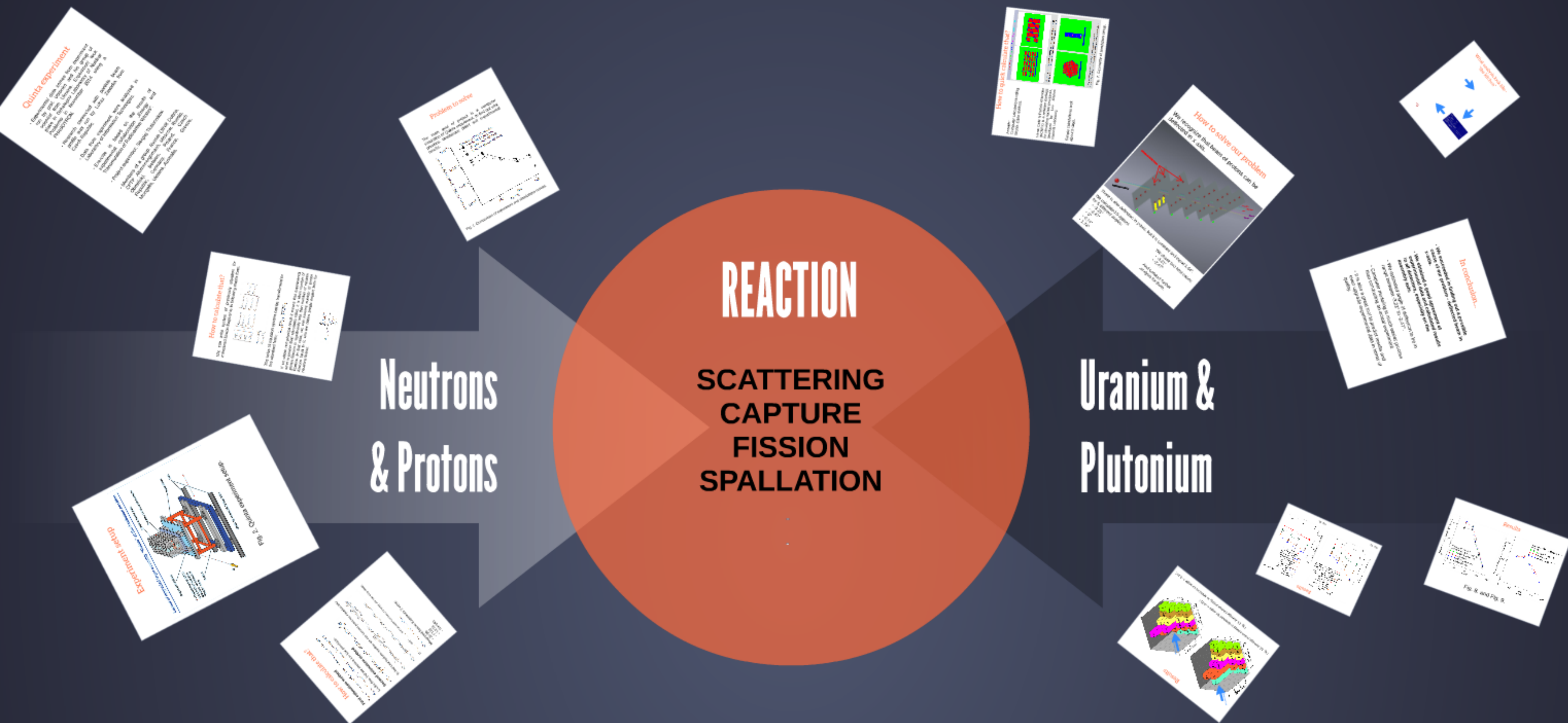


Fission of U-238 and Pu-239 production in subcritical sets

P. Karpowicz, University of Warsaw
M. Konieczny, Warsaw University of Technology
Supervisor:
Andrzej Wojciechowski Ph. D.,
Laboratory of Information Technologies (LIT)



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Problem to solve

The main goal of project is a computer simulation of Quinta experiment, to find out why previous simulations didn't suit experimental results.

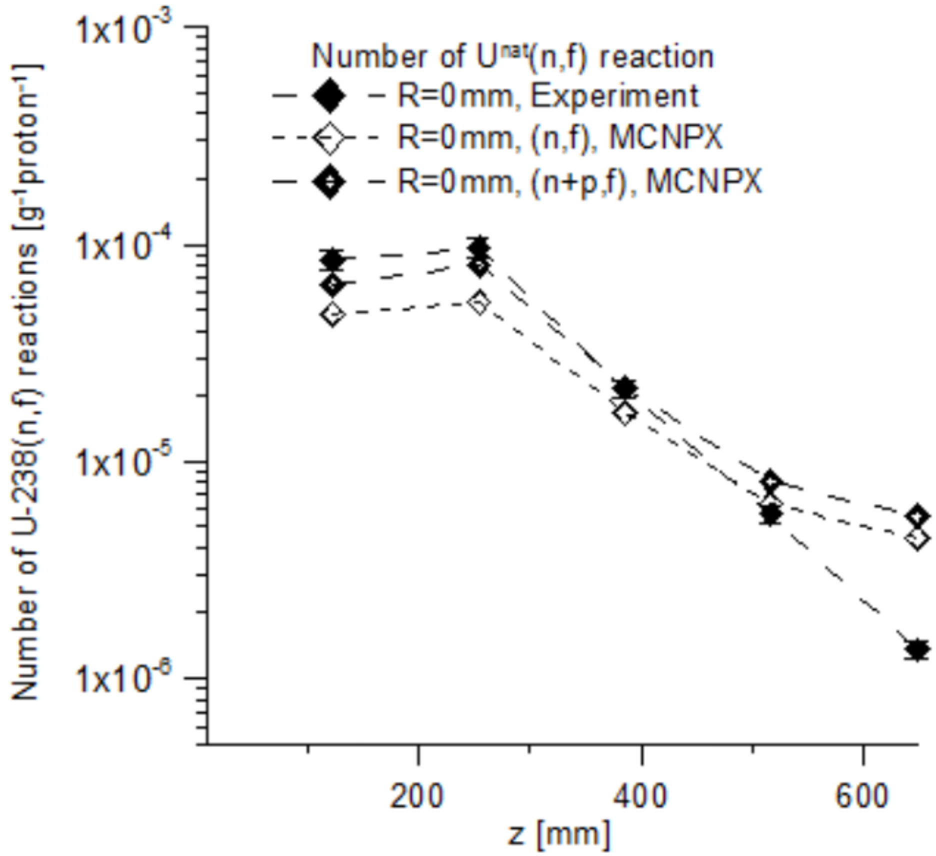


Fig. 1. Comparison of experiment and simulations results.

Quinta experiment

- Experimental data comes from experiment run by prof. Voronko and his group of scientist from Ukraine. Experiment took place in Dzhelepov Laboratory of Nuclear Problems in November 2014 using a PHASOTRON.

- Research connected with particle beam profile was run by Lukas Zavorka from Czech Republic.

- Data from experiment were analyzed in Laboratory of Information Technogies.

- Exercise is based on the results of experimental collaboration „Energy and Transmutation of Radioactive Wastes".

- Project supervisor: Siergiej Tiutiunnikov.

- Members of a group: Russia (JINR Dubna, CPTP Atomenergomash, Moscow, Russia, Obmińsk), Belarus, Poland, Czech Republic, Germany, France, Greece, Mongolia, Ukraine, Australia.

Experiment setup

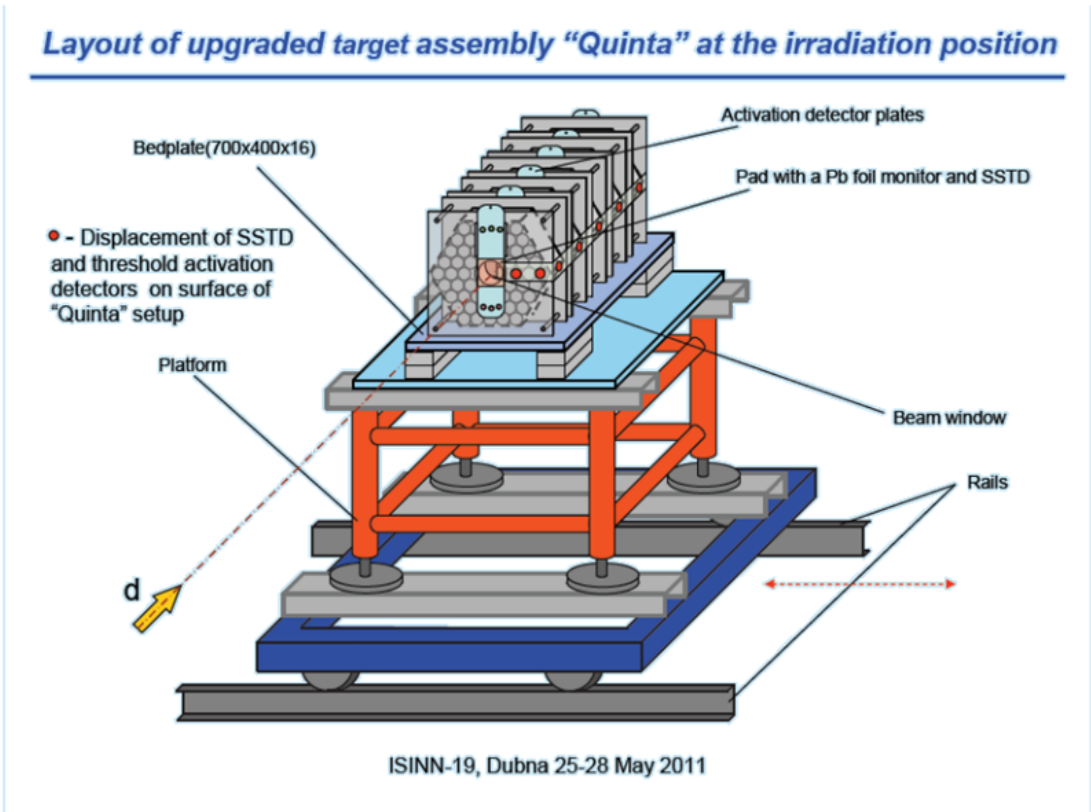


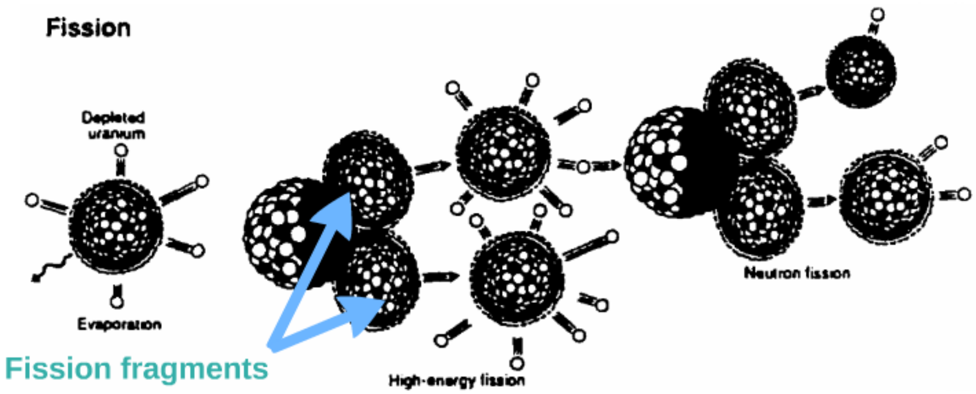
Fig. 2. Quinta experiment setup.

REACTION

SCATTERING
CAPTURE
FISSION
SPALLATION

Fission

Fission is a reaction (decay) of splitting nuclei to lighter parts, with releasing huge amount of energy via neutrons and gamma rays. Reaction can be extorted by neutron flux.

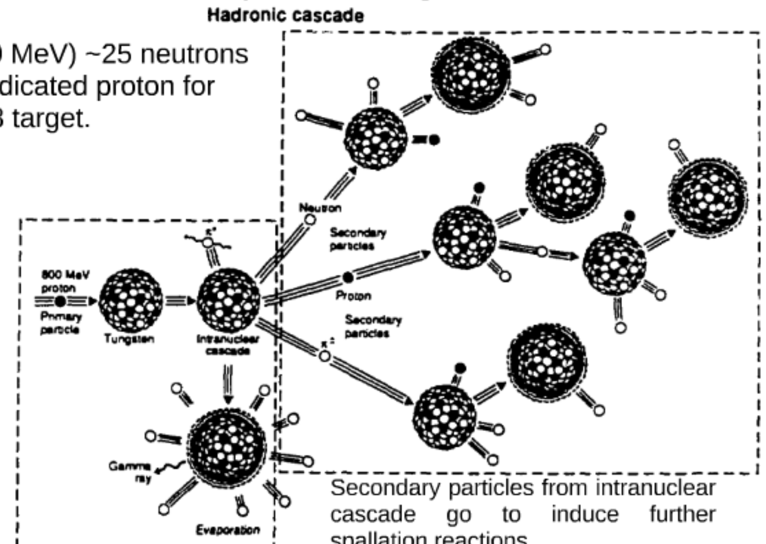


High-energy protons bombarding uranium target, high-energy fission competes with evaporation during de-excitation. More neutrons are emitted from high-energy fission than evaporation. Emitted neutrons can go on to induce more fission events.

Fig. 3. Fission scheme.

Spallation

It's obtained by flux of high energy protons hitting a target (lead, mercury). In effect, nucleus emits several nucleons. Spallation is used to produce high beams of neutrons.



E(660 MeV) ~25 neutrons per indicated proton for U-238 target.

The primary particle hits target, inducting spallation (the intranuclear cascade and evaporation)

Secondary particles from intranuclear cascade go to induce further spallation reactions.

Fig. 4. Spallation scheme.

Fission and spallation happen in the same nucleus!

The cross section for U-238(n,f) reaction sharply increase for energies greater then 1 MeV and peaks for energy equal about 50 MeV. It means that to fission U-238 we must produce neutron with energy greater than 10 MeV.
(Average energy of fission neutrons is equal to ~2 MeV).

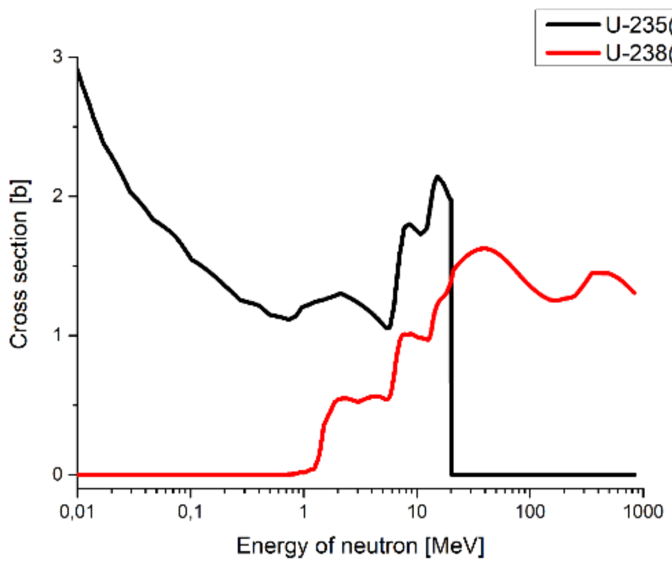


Fig. 5. Cross sections for U-235(n,f) and U-238(n,f).

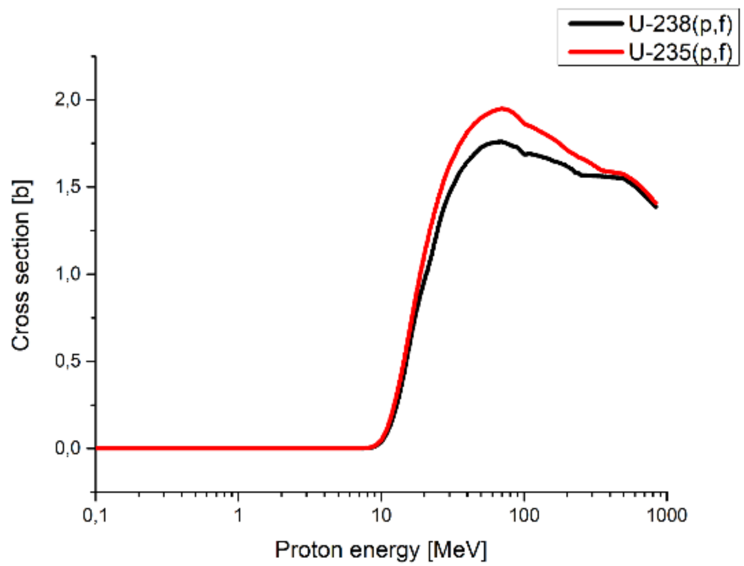


Fig. 6. Cross sections for U-235(p,f) and U-238(p,f).

How to calculate that?

First estimation method

$$Y_{U(n,f)Z} = a_{U235} Y_{U235(n,f)Z}(E_n = 25.3meV) + a_{U238} Y_{U238(n,f)Z}(E_n = 14MeV)$$

$$Y_{U(n,f)Z} = (Y_{U238(n,f)Z}(E_n = 0.5MeV) + Y_{U238(n,f)Z}(E_n = 14MeV))/2$$

Looks fine, but we receive too low precision.

Second estimation method

$$n^{exp}_{U(x,f)Z} = \int_0^\infty (\phi^n(E)\sigma_{U(n,f)Z}(E) + \phi^p(E)\sigma_{U(p,f)Z}(E))\rho\frac{A}{m}dE$$

To find total fissions number we can convert previous equation into:

$$n^{exp}_{U(x,f)Z} = \int_0^\infty (\phi^n(E)\sigma_{U(n,f)}(E)\frac{\sigma_{U(n,f)Z}(E)}{\sigma_{U(n,f)}(E)} + \phi^p(E)\sigma_{U(p,f)}(E)\frac{\sigma_{U(p,f)Z}(E)}{\sigma_{U(p,f)}(E)})\rho\frac{A}{m}dE$$

$$n^{exp}_{U(x,f)Z} = \int_0^\infty (\phi^n(E)\sigma_{U(n,f)}(E)Y_{U(n,f)Z} + \phi^p(E)\sigma_{U(p,f)}(E)Y_{U(p,f)Z})BdE$$

If $Y_{U(p,f)Z}$ and $Y_{U(n,f)Z}$ are constant function of energy one can easy obtain:

$$n^{exp}_{U(x,f)Z} = Y_{U(n,f)Z} n_{U(n,f)} + Y_{U(p,f)Z} n_{U(p,f)}$$

Measured fission fragments Z were:

- Zr-90
- I-131
- I-133
- Ce-143

How to calculate that?

We can write system of previously equation for measured fission fragments in following matrix form:

$$\begin{bmatrix} n^{exp}_{Zr97} \\ n^{exp}_{I131} \\ n^{exp}_{I133} \\ n^{exp}_{Ce143} \end{bmatrix} = \begin{bmatrix} Y_{Zr97,U(n,f)} & Y_{Zr97,U(p,f)} \\ Y_{I131,U(n,f)} & Y_{I131,U(p,f)} \\ Y_{I133,U(n,f)} & Y_{I133,U(p,f)} \\ Y_{Ce143,U(n,f)} & Y_{Ce143,U(p,f)} \end{bmatrix} \begin{bmatrix} n_{U(n,f)} \\ n_{U(p,f)} \end{bmatrix}$$

$$\left.\begin{matrix} n_{U(n,f)} \geq 0 \\ n_{U(p,f)} \geq 0 \end{matrix}\right\}$$

The solve of equation system can be transformed to the standard form:

$$\overline{n} = (\overline{Y^T}\,\overline{Y})^{-1}\,\overline{Y^T}\,\overline{n^{exp}}\,)$$

If we obtain non-physical result it means that experimental error is greater than calculated value. It is when number of proton fissions is significantly less than number of neutron fissions. In that case, we can assume number of proton fissions equal to 0, and we obtain simple explicit form for neutrons fission:

$$n_{U(n,f)} = \frac{\sum_z^4 Y_z n_z^{exp}}{\sum_z^4 Y_z^2}$$

How to quick calculate that?

Answer:
Computer calculations using Monte-Carlo method.

Monte Carlo N-Particle eXtended (MCNPX) is a software package for simulating nuclear processes, developed by Los Alamos National Laboratory.

Simple calculations took about 5 days.

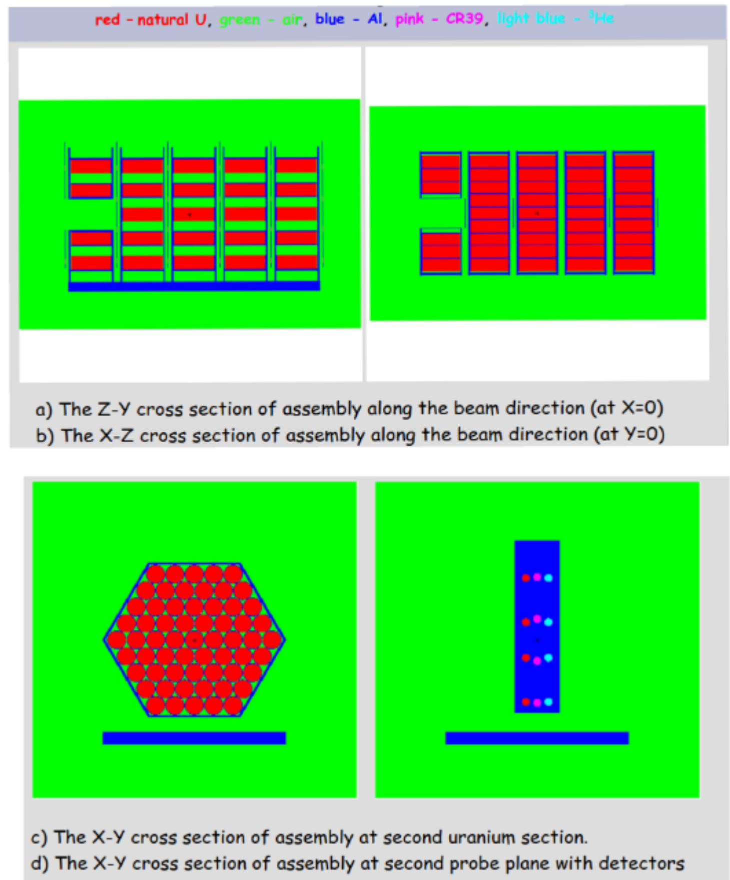
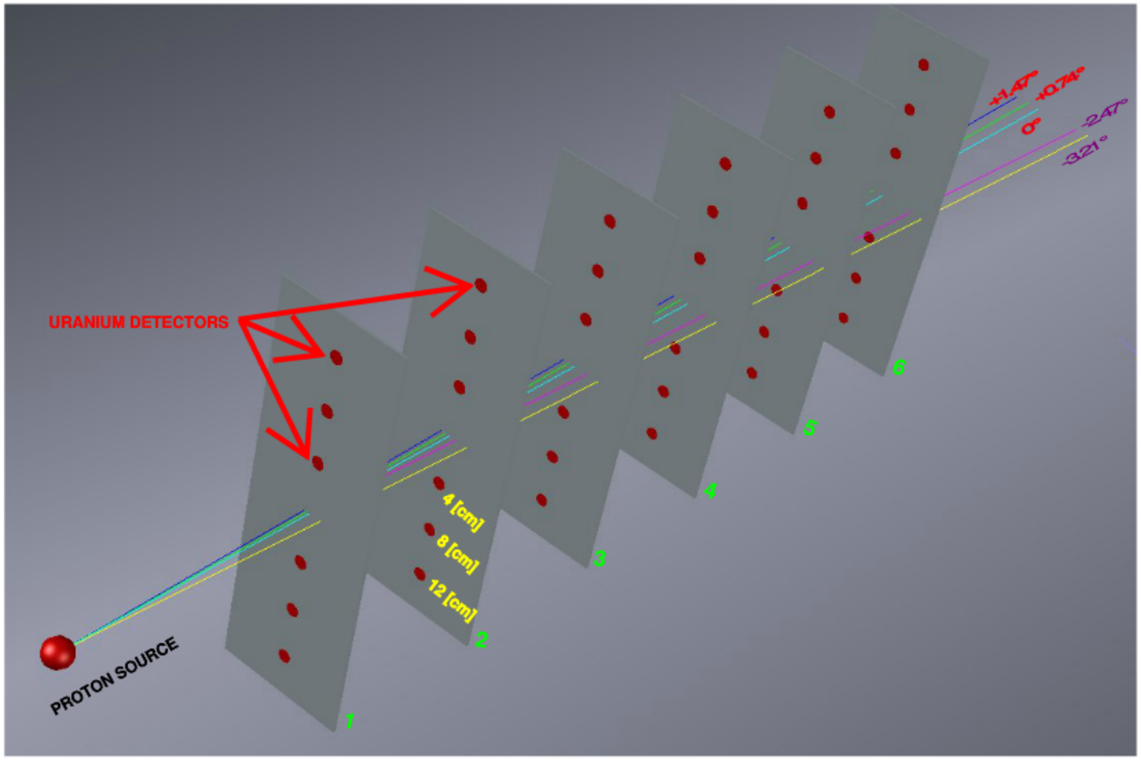


Fig. 7. Geometry of simulation setup.

How to solve our problem

We recognize that beam of protons can be deflected in x axis.



There is also deflection in y-axis, but it is constant and equal 1.84°.

We calculated 5 options for 5 different angles:

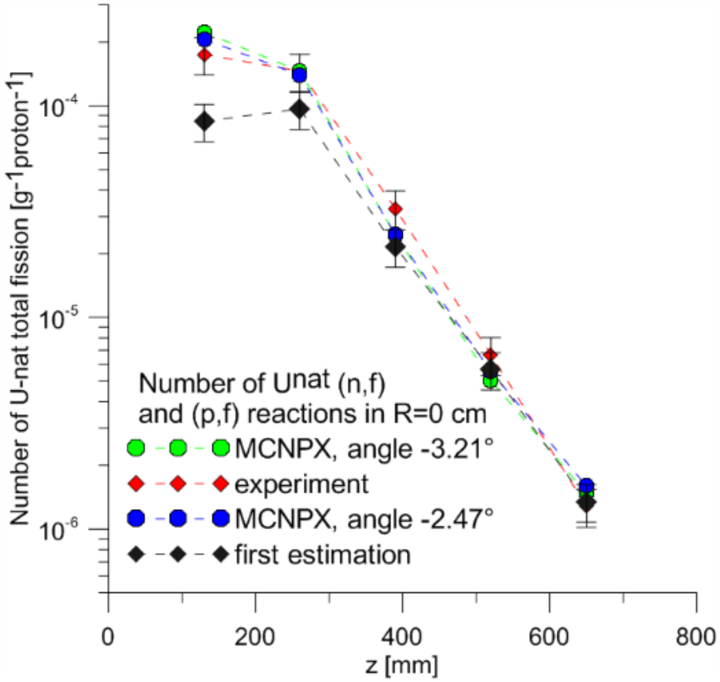
- -3.21°
- -2.47°
- 0°
- 0.74°
- 1.74°

We chose two best cases:

- -3.21°
- -2.47°

And conduct further analysis for them.

Results



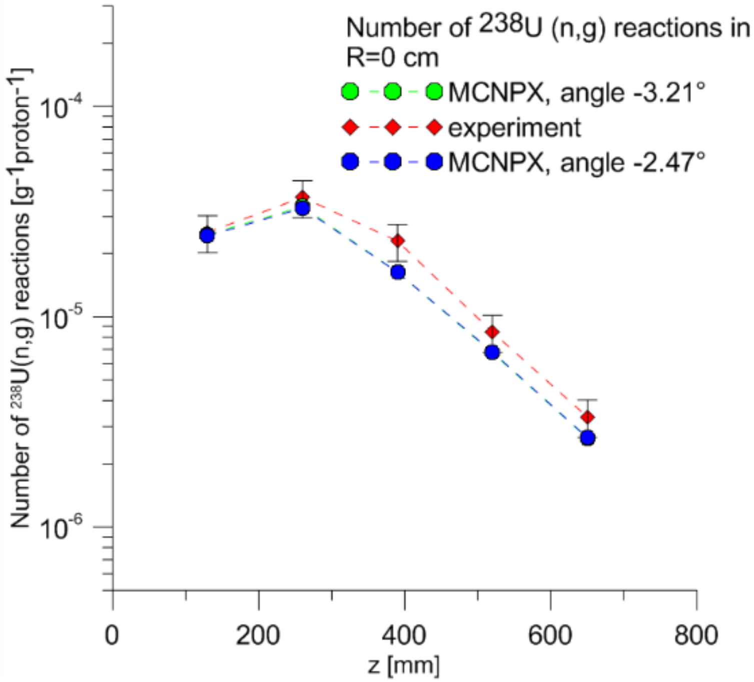


Fig. 8. and Fig. 9.

Results

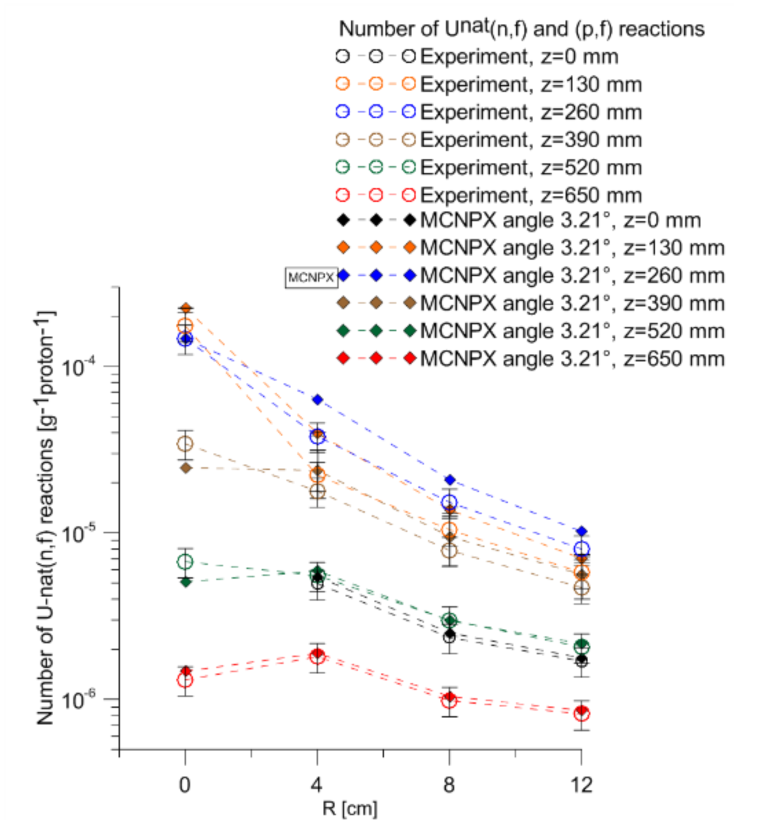


Fig. 10.

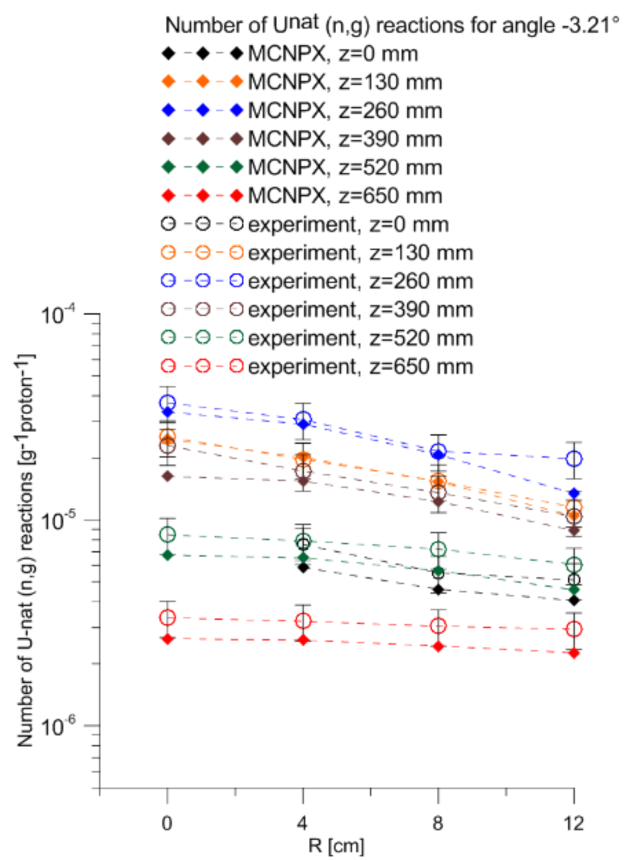


Fig. 11.

Results

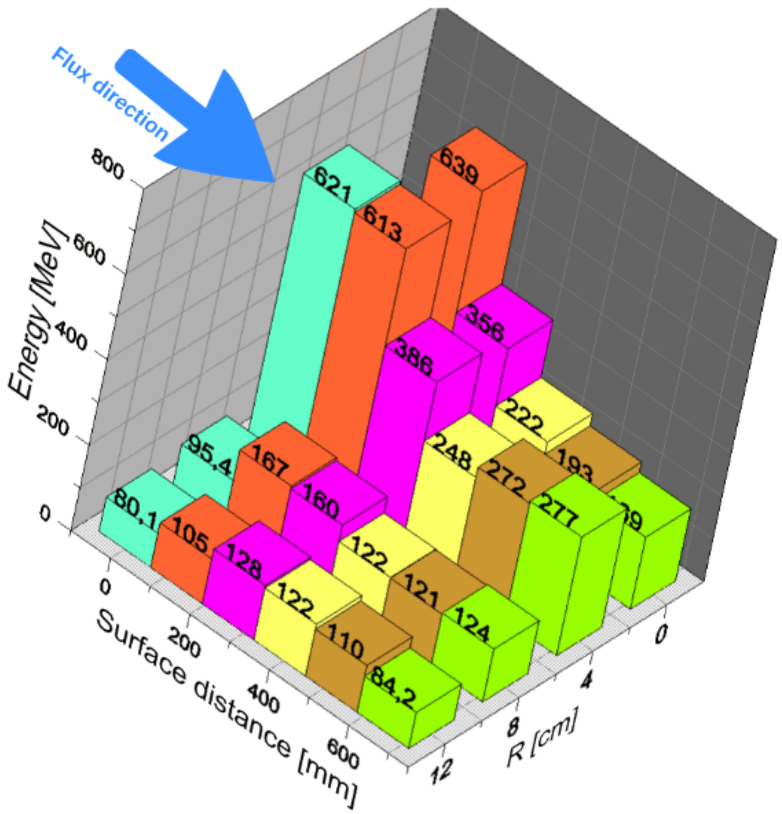


Fig. 12. Average proton energy in detectors for angle = -3.21°.

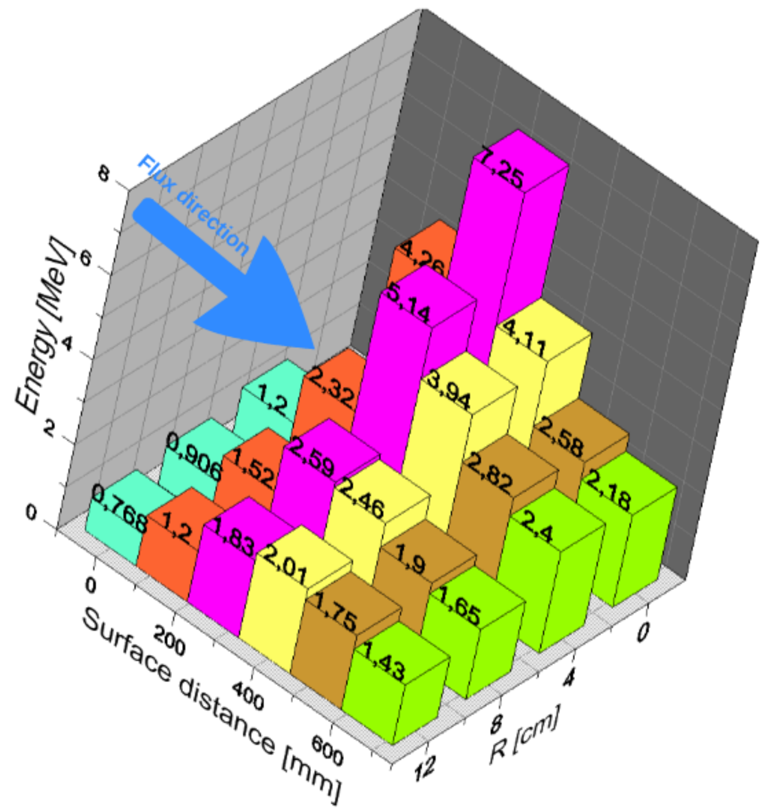
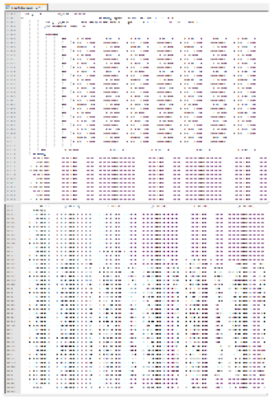


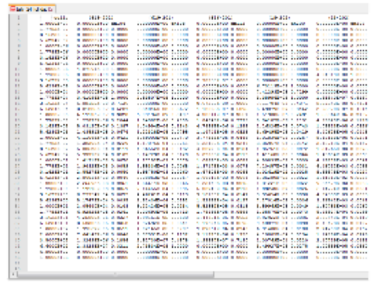
Fig. 13. Average neutron energy in detectors for angle = -3.21°.

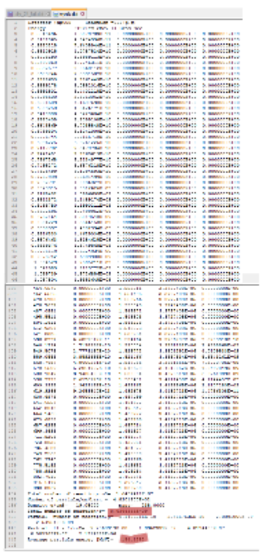
In conclusion...

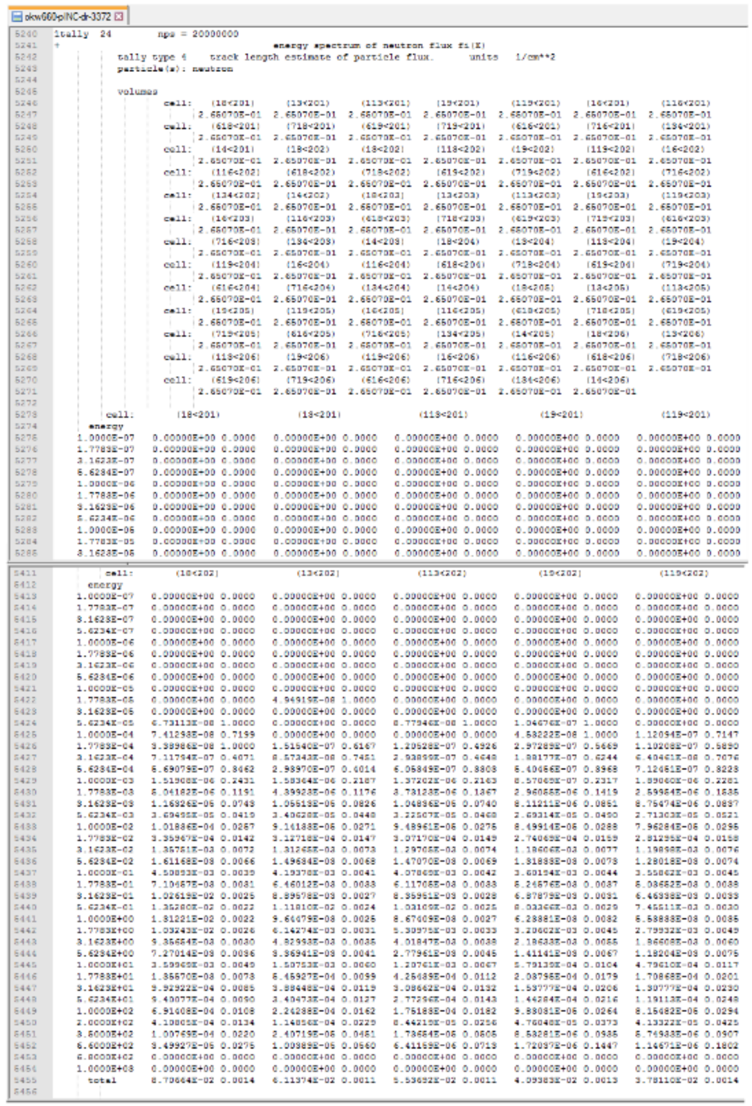
- **We succeeded in finding out a possible cause of our problem - deflected beam in x-axis.**
- **We obtained a good agreement of experimental data and calculated results in all detectors, especially on the assembly axis.**
- We estimated angle of deflection to be in range between -3.21° to -2.47°.
- Computer modeling is much easier process than conducting an actual experiment.
- It is also a great tool to predict results and even upgrade experimental data in terms of quality.

What analysis look like - "the kitchen"









okw660-pINC-dr-3372

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+                                   energy spectrum of neutron flux fi(E)
           tally type 4     track length estimate of particle flux.      units   1/cm**2
           particle(s): neutron

           volumes
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                 cell:  (618<201)    (718<201)    (619<201)    (719<201)    (616<201)    (716<201)    (134<201)
                      2.65070E-01  2.65070E-01  2.65070E-01  2.65070E-01  2.65070E-01  2.65070E-01  2.65070E-01
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   3.1623E-07   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   5.6234E-07   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
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   1.7783E-06   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   3.1623E-06   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
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   1.7783E-05   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   3.1623E-05   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
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```
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     energy
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   3.1623E-07   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   5.6234E-07   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   1.0000E-06   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   1.7783E-06   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   3.1623E-06   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
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   1.7783E-01   7.10487E-03 0.0031   6.46012E-03 0.0033   6.11705E-03 0.0033   5.24676E-03 0.0037   5.08682E-03 0.0038
   3.1623E-01   1.02619E-02 0.0025   8.89578E-03 0.0027   8.35961E-03 0.0028   6.87879E-03 0.0031   6.46338E-03 0.0033
   5.6234E-01   1.35280E-02 0.0022   1.11810E-02 0.0024   1.03109E-02 0.0025   8.03366E-03 0.0029   7.45511E-03 0.0030
   1.0000E+00   1.31221E-02 0.0022   9.64479E-03 0.0026   8.67409E-03 0.0027   6.28881E-03 0.0032   5.58888E-03 0.0036
   1.7783E+00   1.03243E-02 0.0026   6.14274E-03 0.0031   5.30975E-03 0.0033   3.20602E-03 0.0045   2.79932E-03 0.0049
   3.1623E+00   9.36654E-03 0.0030   4.82998E-03 0.0035   4.01847E-03 0.0038   2.18633E-03 0.0055   1.86608E-03 0.0060
   5.6234E+00   7.27014E-03 0.0036   3.36941E-03 0.0041   2.77961E-03 0.0045   1.41141E-03 0.0067   1.18204E-03 0.0075
   1.0000E+01   3.59969E-03 0.0049   1.50753E-03 0.0060   1.20761E-03 0.0067   5.79139E-04 0.0104   4.79610E-04 0.0117
   1.7783E+01   1.86670E-03 0.0078   5.45927E-04 0.0099   4.25489E-04 0.0112   2.08798E-04 0.0179   1.70868E-04 0.0201
   3.1623E+01   9.92922E-04 0.0085   3.88448E-04 0.0119   3.08662E-04 0.0132   1.53777E-04 0.0206   1.30777E-04 0.0230
   5.6234E+01   9.40077E-04 0.0090   3.40473E-04 0.0127   2.77296E-04 0.0143   1.44284E-04 0.0216   1.19113E-04 0.0248
   1.0000E+02   6.91408E-04 0.0108   2.24288E-04 0.0162   1.75188E-04 0.0182   9.88081E-05 0.0264   8.15482E-05 0.0294
   2.0000E+02   4.10805E-04 0.0134   1.14856E-04 0.0229   8.44215E-05 0.0256   4.76048E-05 0.0373   4.13322E-05 0.0425
   3.6000E+02   1.00769E-04 0.0220   2.40719E-05 0.0481   1.73684E-05 0.0605   8.58281E-06 0.0935   5.74938E-06 0.0907
   6.6000E+02   3.49927E-05 0.0275   1.00989E-05 0.0560   6.41169E-06 0.0719   1.72097E-06 0.1447   1.14671E-06 0.1802
   6.8000E+02   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
   1.0000E+03   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000   0.00000E+00 0.0000
     total      8.70664E-02 0.0014   6.11374E-02 0.0011   5.53692E-02 0.0011   4.09383E-02 0.0013   3.78110E-02 0.0014
```

tally_24_full.dat

cell:	(618<201)		(619<201)		(616<201)		(18<202)		(13<202)	
1.0000E-07	0.00000E+00	dn104	0.00000E+00	dn108	0.00000E+00	dn112	0.00000E+00	dn200	0.00000E+00	dn204
1.7783E-07	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
3.1623E-07	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
5.6234E-07	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
1.0000E-06	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
1.7783E-06	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
3.1623E-06	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
5.6234E-06	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
1.0000E-05	4.87691E-08	1.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
1.7783E-05	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	4.94919E-08	1.0000
3.1623E-05	0.00000E+00	0.0000	0.00000E+00	0.0000	2.38358E-07	1.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
5.6234E-05	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	6.73113E-08	1.0000	0.00000E+00	0.0000
1.0000E-04	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	7.41293E-08	0.7199	0.00000E+00	0.0000
1.7783E-04	3.41096E-08	1.0000	2.53507E-08	1.0000	0.00000E+00	0.0000	3.38986E-08	1.0000	1.51540E-07	0.6167
3.1623E-04	6.88155E-08	0.7136	0.00000E+00	0.0000	9.67260E-09	1.0000	7.11794E-07	0.4071	8.57343E-08	0.7451
5.6234E-04	3.81885E-07	0.5287	9.34046E-08	0.5672	1.18023E-07	0.6095	5.69079E-07	0.3462	2.93970E-07	0.4014
1.0000E-03	6.17136E-07	0.3020	6.89194E-07	0.3487	4.38011E-07	0.3551	1.51908E-06	0.2431	1.58364E-06	0.2187
1.7783E-03	1.22829E-06	0.2044	1.54082E-06	0.2305	1.84645E-06	0.2225	5.04182E-06	0.1191	4.39923E-06	0.1176
3.1623E-03	5.03120E-06	0.1387	4.04994E-06	0.1343	4.76969E-06	0.1288	1.16326E-05	0.0743	1.05513E-05	0.0826
5.6234E-03	1.43591E-05	0.0674	1.22328E-05	0.0795	1.14741E-05	0.0828	3.69495E-05	0.0419	3.40628E-05	0.0448
1.0000E-02	4.11069E-05	0.0421	3.69003E-05	0.0470	3.77853E-05	0.0449	1.01836E-04	0.0257	9.14133E-05	0.0271
1.7783E-02	1.38390E-04	0.0229	1.17076E-04	0.0251	1.22122E-04	0.0247	3.35967E-04	0.0142	3.12718E-04	0.0147
3.1623E-02	5.00523E-04	0.0117	4.36552E-04	0.0132	4.00728E-04	0.0134	1.35751E-03	0.0072	1.31265E-03	0.0073
5.6234E-02	5.93318E-04	0.0109	5.14723E-04	0.0120	5.09335E-04	0.0119	1.61168E-03	0.0066	1.49634E-03	0.0068
1.0000E-01	1.41215E-03	0.0070	1.17152E-03	0.0079	1.08690E-03	0.0083	4.50893E-03	0.0039	4.19378E-03	0.0041
1.7783E-01	1.94329E-03	0.0059	1.55336E-03	0.0068	1.37173E-03	0.0073	7.10457E-03	0.0031	6.46012E-03	0.0033
3.1623E-01	2.55279E-03	0.0050	1.95750E-03	0.0060	1.67605E-03	0.0065	1.02619E-02	0.0025	8.89578E-03	0.0027
5.6234E-01	2.86914E-03	0.0046	2.08853E-03	0.0057	1.66918E-03	0.0064	1.35280E-02	0.0022	1.11810E-02	0.0024
1.0000E+00	2.25592E-03	0.0051	1.56614E-03	0.0065	1.23149E-03	0.0075	1.31221E-02	0.0022	9.64479E-03	0.0025
1.7783E+00	1.19115E-03	0.0067	7.14455E-04	0.0095	5.30740E-04	0.0113	1.03243E-02	0.0026	6.14274E-03	0.0031
3.1623E+00	8.29354E-04	0.0079	4.46261E-04	0.0119	3.10133E-04	0.0142	9.35654E-03	0.0030	4.82993E-03	0.0035
5.6234E+00	5.17473E-04	0.0095	2.51436E-04	0.0152	1.52653E-04	0.0197	7.27014E-03	0.0036	3.36941E-03	0.0041
1.0000E+01	2.05440E-04	0.0143	9.22426E-05	0.0234	5.52992E-05	0.0318	3.59969E-03	0.0049	1.50753E-03	0.0060
1.7783E+01	6.92929E-05	0.0261	3.00693E-05	0.0413	1.75496E-05	0.0569	1.35570E-03	0.0073	5.45927E-04	0.0099
3.1623E+01	4.25800E-05	0.0327	1.99151E-05	0.0500	1.44751E-05	0.0653	9.92922E-04	0.0085	3.88448E-04	0.0119
5.6234E+01	3.60306E-05	0.0368	1.92692E-05	0.0541	1.17624E-05	0.0695	9.40077E-04	0.0090	3.40473E-04	0.0127
1.0000E+02	1.97567E-05	0.0511	1.02165E-05	0.0787	7.87361E-06	0.0940	6.91408E-04	0.0108	2.24238E-04	0.0162
2.0000E+02	7.64149E-06	0.0846	5.07031E-06	0.1102	3.11370E-06	0.1390	4.10805E-04	0.0134	1.14856E-04	0.0229
3.5000E+02	1.32496E-06	0.1856	2.51730E-07	0.4578	1.95587E-07	0.7130	1.00769E-04	0.0220	2.40719E-05	0.0451
6.6000E+02	2.33238E-07	0.3211	2.73834E-08	1.0000	0.00000E+00	0.0000	3.49927E-05	0.0275	1.00389E-05	0.0560
6.8000E+02	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000
1.0000E+03	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000	0.00000E+00	0.0000

```
 numer kolumny:
3
 numer kolumny          3
  Name of input file:        flux.dat

    Name of detector:          dp608
Number of flux data     45
  Name of input file:        sigma.dat

    Name of detector:          238(p.f)
Number of cross sect    61
 Name of output file:        wynik.dat

 Total number of reaction/cm3=   1.6452411E-07
 Total number of reaction/g=  8.6364365E-09
 Partial  number of reaction/g=  1.1486316E-10    3.4557635E-10    2.0891096E-09
    2.5495490E-09
 Partial  flux 1/cm2g=  7.2968319E-08    1.0362331E-07    4.8194488E-07
  6.5853652E-07    2.1912888E-06
 Average particle energy [MeV]=   154.1252
```

tally_24_full.dat | wynik.dat

```
Detector :dp608     Reaction =238(p.f)
Energy             Flux[cm-2MeV-1] Cros.sec
 0.5043436         1.5070703E-11  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5131065         4.5474336E-11  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5220215         7.6405944E-11  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5310915         1.0787524E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5403190         1.3989113E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5497068         1.7246311E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5592577         2.0560112E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5689746         2.3931507E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5788603         2.7361449E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5889178         3.0851016E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.5991501         3.4401215E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6095600         3.8013062E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6201509         4.1687698E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6309258         4.5426171E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6418880         4.9229604E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6530405         5.3099108E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6643869         5.7035848E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6759304         6.1041000E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6876744         6.5115718E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.6996225         6.9261225E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7117782         7.3478784E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7241451         7.7769613E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7367268         8.2134977E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7495272         8.6576218E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7625499         9.1094593E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7757989         9.5691499E-10  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.7892781         1.0036827E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8029915         1.0512626E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8169432         1.0996695E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8311372         1.1489174E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8455780         1.1990211E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8602695         1.2499952E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8752164         1.3018548E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.8904229         1.3546158E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9058937         1.4082934E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9216333         1.4629036E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9376463         1.5184625E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9539376         1.5749868E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9705119         1.6324934E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
 0.9873742         1.6909988E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
  1.004529         1.7505211E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
  1.021983         1.8110772E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
  1.039739         1.8726856E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
  1.057804         1.9353645E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
```

```
   1.039739     1.8726856E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
   1.057804     1.9353645E-09  0.0000000E+00  0.0000000E+00  0.0000000E+00
```

```
   454.6570     0.0000000E+00   1.552211      3.3727408E-06  0.0000000E+00
   462.5565     0.0000000E+00   1.551495      3.3727408E-06  0.0000000E+00
   470.5933     0.0000000E+00   1.550766      3.3727408E-06  0.0000000E+00
   478.7697     0.0000000E+00   1.550025      3.3727408E-06  0.0000000E+00
   487.0881     0.0000000E+00   1.549271      3.3727408E-06  0.0000000E+00
   495.5511     0.0000000E+00   1.548503      3.3727408E-06  0.0000000E+00
   504.1611     0.0000000E+00   1.546223      3.3727408E-06  0.0000000E+00
   512.9207     0.0000000E+00   1.542273      3.3727408E-06  0.0000000E+00
   521.8325     0.0000000E+00   1.538254      3.3727408E-06  0.0000000E+00
   530.8992     6.6851517E-11   1.534164      3.3736787E-06  9.3789310E-10
   540.1234     1.7138398E-10   1.530004      3.3761182E-06  2.4395868E-09
   549.5078     2.7773137E-10   1.525772      3.3801291E-06  4.0109533E-09
   559.0553     3.8592693E-10   1.521466      3.3857834E-06  5.6543161E-09
   568.7687     4.9600241E-10   1.517085      3.3931556E-06  7.3720705E-09
   578.6509     5.2064797E-10   1.512628      3.4010052E-06  7.8497049E-09
   588.7047     3.9405498E-10   1.508094      3.4070313E-06  6.0261809E-09
   598.9333     2.6526176E-10   1.503481      3.4111458E-06  4.1144457E-09
   609.3395     1.3423132E-10   1.498162      3.4132565E-06  2.1107183E-09
   619.9266     9.2453540E-13   1.492678      3.4132713E-06  1.4736308E-11
   630.6976     0.0000000E+00   1.487099      3.4132713E-06  0.0000000E+00
   641.6557     0.0000000E+00   1.481422      3.4132713E-06  0.0000000E+00
   652.8042     0.0000000E+00   1.475647      3.4132713E-06  0.0000000E+00
   664.1465     0.0000000E+00   1.469772      3.4132713E-06  0.0000000E+00
   675.6858     0.0000000E+00   1.463795      3.4132713E-06  0.0000000E+00
   687.4255     0.0000000E+00   1.457714      3.4132713E-06  0.0000000E+00
   699.3693     0.0000000E+00   1.451527      3.4132713E-06  0.0000000E+00
   711.5206     0.0000000E+00   1.445774      3.4132713E-06  0.0000000E+00
   723.8830     0.0000000E+00   1.439951      3.4132713E-06  0.0000000E+00
   736.4602     0.0000000E+00   1.434027      3.4132713E-06  0.0000000E+00
   749.2560     0.0000000E+00   1.428000      3.4132713E-06  0.0000000E+00
   762.2740     0.0000000E+00   1.421869      3.4132713E-06  0.0000000E+00
   775.5182     0.0000000E+00   1.415631      3.4132713E-06  0.0000000E+00
   788.9926     0.0000000E+00   1.409284      3.4132713E-06  0.0000000E+00
   802.7010     0.0000000E+00   1.402841      3.4132713E-06  0.0000000E+00
   816.6477     0.0000000E+00   1.396342      3.4132713E-06  0.0000000E+00
   830.8367     0.0000000E+00   1.389730      3.4132713E-06  0.0000000E+00
 Total number of reaction/cm3=  1.6452411E-07
 Number of particle/cm*barn =  4.8201300E-02
 Density g/cm3=   19.05000       masa=   238.0000
 Total number of reaction/g=  8.6364365E-09
 Partial  number of reaction/g=  1.1486316E-10    3.4557635E-10    2.0891096E-09
   2.5495490E-09
 Partial  flux 1/cm2=  7.2968319E-08    1.0362331E-07    4.8194488E-07
  6.5853652E-07    2.1912888E-06
 Average particle energy [MeV]=   154.1252
```

THANK YOU!

And we hope we resolved all your doubts

Prezi

Sources

- Spallation Physics - An Overview, Gary J. Russel
- Neutron Generation, John M. Carpenter
- Users manual for MCNPX 2.7.0