Average high energy neutron flux distribution in the Quinta sub-critical assembly irradiated with proton beam of 0.66 GeV energy applying the actinide spectral index method.

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1. Introduction

1.1. Fast neutron fluency measurement methods.

- The fast neutron fluence measurement method consists in utilizing neutron irradiated actinide samples for estimating neutron fluence and average neutron energy inside the volume of samples.

- The idea of the actinide spectral index method is to search the neutron energy \(E_d\) for the ratio \(\alpha(E_d)\) of fission cross section \(\sigma_f(E_d)\) to capture cross section \(\sigma_c(E_d)\) of the selected actinide isotope from the nuclear data base that is equal to the measured ratio \(\alpha_m\) of fissioned \((N_{yf})\) and captured \((N_{yc})\) actinide isotopes (spectral indexes) [1, 2]:

\[
\alpha(E_d) = \frac{\sigma_f(E_d)}{\sigma_c(E_d)} = \alpha_m = \frac{N_{yf}}{N_{yc}} = \frac{\bar{\sigma}_f}{\bar{\sigma}_c}
\]
1. Introduction

1.1. Fast neutron fluency measurement methods.

- It is useful to look closely at the ratios $\alpha = \sigma_f / \sigma_c$ of the capture and fission cross section of the Np-237 isotope.
- The fission/absorption ratios are consistently higher for the fast-neutron spectrum. Thus, in a fast spectrum, actinides are preferentially fissioned, not transmuted into higher actinides.

Fig. 1a. Cross-sections of Np-237(n,g)Np-238 and Np-237(n,f) reactions.
1. Introduction

1.1. Fast neutron fluency measurement methods.

R = 60 cm, L= 110 cm, $k_{eff} \approx 0.23$


Most of the neutrons in the neutron spectrum generated in the uranium target are concentrated in the energy range from tens of keV to ~ 100 MeV

$^{238}$U(n,f) – neutron energy > 1 MeV

The neutron spectrum also contains a fast component, which is essential for transmutation via fast fission and other reactions.

Neutron Energy, MeV

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1. Introduction

1.1. Fast neutron fluency measurement methods.

Since the measured spectral indexes ($\alpha_m$) is defined as the ratio of average fission ($\bar{\sigma}_f$) and capture ($\bar{\sigma}_c$) cross sections so the ratio ($\alpha(E_d)$) of retrieved distinct fission ($\sigma_f(E_d)$) and capture ($\sigma_c(E_d)$) cross sections for the distinct neutron energy ($E_d$) from the nuclear data base describe the average values:

$$E_d = \bar{E} ; \quad \sigma_f(E_d) = \bar{\sigma}_f ; \quad \sigma_c(E_d) = \bar{\sigma}_c$$

Having the average fission and capture cross section values we can evaluate the average neutron flux ($\bar{\phi}$) in the location of the actinide sample using the measured amount of fissioned and captured actinide isotopes.
1. Introduction

1.1. Fast neutron fluency measurement methods.

• The amount of neutron induced fissioned ($N_{yf}$) and neutron captured actinide isotopes ($N_{yc}$) in the actinide sample of volume $V_p$ can be expressed:

\[
N_{yf} = V_p \bar{\Phi} N \bar{\sigma}_f t \tag{1}
\]

\[
N_{yc} = V_p \bar{\Phi} N \bar{\sigma}_c t \tag{2}
\]

Where

$V_p$ - actinide sample volume [cm$^3$],
$\bar{\Phi}$ - average neutron flux in the place of actinide sample location [n/cm$^2$·s],
$N$ – number of actinide isotopes in volume unit [cm$^{-3}$],
$\bar{\sigma}_f$ ; $\bar{\sigma}_c$ -average microscopic cross section for the reactions $(n, f)$ and $(n, \gamma)$ respectively [barns],
$t$ - irradiation time [s].

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1. Introduction

1.1. Fast neutron fluency measurement methods.

- Two different equations for fissioned ($N_{yf}$) and captured ($N_{yc}$) actinide isotopes should give the same average neutron flux value what is a proof for correct measurements.

\[
\frac{N_{yf}}{N_{yc}} = \frac{V_p \bar{\phi} N \bar{\sigma}_f t}{V_p \bar{\phi} N \bar{\sigma}_c t} = \frac{\bar{\sigma}_f}{\bar{\sigma}_c}
\]
1. Introduction

1.2. Metallic natural uranium as activation detector foil

- Since metallic natural uranium consists of uranium-238 (99.2752%), uranium-235 (0.7202%), and a very small amount of uranium-234 so the irradiated detector foil introduces an additional error in the measurement of the average neutron flux and neutron fluency.

- Since the measurements of the amount of fissions in the irradiated natural uranium foil constitute the sum of U-238 and U-235 fissions, Eq. (1) must be modified. In contrast, the measurement of neutron captures is based on the measurement of the amount of Pu-239 produced (see Eq.(3)). So the Eq. (2) does not have to be modified since neutron captures are not taken into (considered) account by U-235.

\[
^{238}\text{U(n,}\gamma^{239}\text{U}} \to ^{239}\beta^{239}\text{Np} \to ^{239}\beta^{239}\text{Pu}
\]

(3)

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1. Introduction

1.2. Metallic natural uranium as activation detector foil

In order to include this we take into account the number of neutron fission induced \(N_{yfs}\) which is the sum of U-238 \(N_{yf8}\) and U-235 \(N_{yf5}\) fissions and neutron captured actinide isotopes \(N_{yc8}\) in the actinide foil of volume \(V_p\) what is expressed by the following equations:

\[
N_{yfs} = N_{yf8} + N_{yf5} = V_p \bar{\phi} N_8 \bar{\sigma}_{f8} t + V_p \bar{\phi} N_5 \bar{\sigma}_{f5} t = V_p \bar{\phi} N_8 t (\bar{\sigma}_{f8} + \frac{N_5}{N_8} \bar{\sigma}_{f5}) \tag{4}
\]

\[
N_{yc8} = V_p \bar{\phi} N_8 \bar{\sigma}_{c8} t \tag{5}
\]

Where,

\(N_8\) – number of U-238 atoms in volume unit of actinide foil [cm\(^{-3}\)],

\(N_5\) – number of U-235 atoms in volume unit of actinide foil [cm\(^{-3}\)],

\(\bar{\sigma}_{f8}\) U-238 average microscopic cross section for the reactions (n, f) [barns],

\(\bar{\sigma}_{f5}\) U-235 average microscopic cross section for the reactions (n, f) [barns],

\(\bar{\sigma}_{c8}\) U-238 average microscopic cross section for the reactions (n, \(\gamma\)) [barns],

\(\bar{\phi}\) - average neutron flux in the place of actinide sample location [n/cm\(^2\)∙s],

\[
N_5/ N_8 = 0.7202/99.2752 = 0.00725 = 7.25 \times 10^{-3}
\]
1. Introduction
1.2. Metallic natural uranium as activation detector foil

The quotient of equations 4 and 5 gives the measured spectral index of the irradiated sample on the left, and on the right we get the expression ($\alpha_{m85}$) which becomes equal to the measured index when we find the neutron energy applying try and error method from the data base for which the relevant fission and capture cross section of the U-238 and U-235 fulfill the equation.

$$\alpha_{m85}(E_d) = \frac{N_{yfs}}{N_{yc8}} = \frac{\bar{\sigma}_{f8}}{\bar{\sigma}_{c8}} + \frac{N_5 \bar{\sigma}_{f5}}{N_8 \bar{\sigma}_{c8}}$$  \hspace{1cm} (6)
1. Introduction
1.2. Metallic natural uranium as activation detector foil

Another words the idea of the method is to search the neutron energy \( (E_d) \) for the ratio \( (\alpha_{m85}(E_d)) \) of fission cross section \( (\sigma_{f8}(E_d)) \) to capture cross section \( (\sigma_{c8}(E_d)) \) of the selected actinide isotope U-238 plus the ratio of fission cross section \( (\sigma_{f5}(E_d)) \) to capture cross section \( (\sigma_{c5}(E_d)) \) of the selected actinide isotope U-235 multiplied by \( N_5/ N_8 = 0.7202/99.2752 = 7.25 \times 10^{-3} \) from the nuclear data base that is equal to the measured ratio \( (\alpha_{m85}(E_d)) \) of fissioned \( (N_{yfs}) \) and captured \( (N_{yc8}) \) actinide isotopes:

\[
\alpha_{m85}(E_d) = \frac{N_{yfs}}{N_{yc8}} = \frac{\bar{\sigma}_{f8}}{\bar{\sigma}_{c8}} + 7.25 \times 10^{-3} \frac{\bar{\sigma}_{f5}}{\bar{\sigma}_{c8}} \quad (7)
\]
2. Experimental part.
2.1. Subcritical assembly Quinta

Fig. 1. Schema of Quinta assembly. On the left there is a view on the uranium target with supporting structures and plastics used for sample placement (detector’s plates), on the right there is a view on the lead shielding enfolding the target with marked the transmutation samples box (window) for the actinides sample location in the shielding.
2. Experimental part.

2.1. Subcritical assembly Quinta

- Dimensions of Quinta assembly.
2. Experimental part.
2.1. Subcritical assembly Quinta

The layout of the uranium foils location on the detector plate. Each plate have 5 positions at the different distances.

Uranium detectors were fixed on the detector plates in dependence on the distance from primary beam axes – 0, 4, 8 and 12 cm.
The dimensions of the foils – diameter 8 mm, thickness 1 mm, weight 1 g.
2. Experimental part.
2.2. Location of activation detector foils in the Quinta sub-critical assembly.

- The location coordinates of all of 23 uranium detectors are shown in Table 1 relative to the axis of the target (along the radius $R$ of the uranium target and along the axis of the target $Z$).

Table 1. The location coordinates of all of 23 uranium detectors.

<table>
<thead>
<tr>
<th>$R /Z$, mm</th>
<th>Foil plates</th>
</tr>
</thead>
<tbody>
<tr>
<td>$R$ - vertically</td>
<td>$Z$ - horizontally</td>
</tr>
<tr>
<td>1</td>
<td>2</td>
</tr>
<tr>
<td>0</td>
<td>123</td>
</tr>
<tr>
<td>$U_{01}$</td>
<td>$U_{11}$</td>
</tr>
<tr>
<td>40</td>
<td>$U_{02}$</td>
</tr>
<tr>
<td>$U_{03}$</td>
<td>$U_{13}$</td>
</tr>
<tr>
<td>120</td>
<td>$U_{04}$</td>
</tr>
</tbody>
</table>
2. Experimental part.

2.3. Measurement.

2.3.1. Irradiation details.

- The Quinta target was irradiated with a pulsed proton beam of 0.66 GeV energy extracted from the Phasotron accelerator, located at the JINR.
- Total number of protons of the irradiation is equal to $8.5 \times 10^{14}$ during the time of irradiation equal to **20580 seconds** (5h 43 min).
- Prior to the irradiation, several polaroid films were placed on the front of Quinta to ensure the proton beam was striking in the centre of the beam window.
2. Experimental part.
2.3. Measurement.
2.3.2. Results

- After the end of irradiation, the uranium foils were taken out from the target to measure γ-spectra using HPGe detectors. Measurement of gamma-ray spectra of irradiated foils was performed in 4 hours after the end of irradiation (more than 10 half-lives of $^{239}$U). In this period 99.9% of $^{239}$U nuclei have decayed to $^{239}$Np.

- The number of fissions was determined by yield of gamma-lines 743.36 keV (93%), 364.49 keV (81.5%), 529.9 keV (87%), and 293.3 keV (42.8%) of fission fragments $^{97}$Zr – 5.7%, $^{131}$I – 3.6%, $^{133}$I – 6.3%, $^{143}$Ce – 4.3%. respectively.

- The number of neutron radiation capture reactions was determined by the yield of -line with energy of 277.6 keV γ-line (I=14.44%) accompanying decay of 239Np (see Eq. 3) [3, 4].
2. Experimental part.

2.3. Measurement.

2.3.2. Results

- The tables below (Table 2 and Table 3,) show the results of measurements of the fission numbers and the number of capture reactions per 1 deuteron and per 1 g of natural uranium.

Table 2. Fission number $N \times 10^{-5}$, fiss/g/p

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>$R/Z$ mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>4.24</td>
<td>8.66</td>
<td>2.22</td>
<td>0.56</td>
<td>0.14</td>
</tr>
<tr>
<td>40</td>
<td>0.53</td>
<td>2.36</td>
<td>3.67</td>
<td>1.65</td>
<td>0.51</td>
<td>0.14</td>
</tr>
<tr>
<td>80</td>
<td>0.25</td>
<td>1.09</td>
<td>1.61</td>
<td>0.82</td>
<td>0.31</td>
<td>0.10</td>
</tr>
<tr>
<td>120</td>
<td>0.18</td>
<td>0.61</td>
<td>0.83</td>
<td>0.49</td>
<td>0.22</td>
<td>0.09</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

- The tables below (Table 2 and Table 3,) show the results of measurements of the fission numbers and the number of capture reactions per 1 deuteron and per 1 g of natural uranium.

Table 3. Capture number, \( N \times 10^{-5}, \text{239Pu/g/p} \)

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/Z mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>2.57</td>
<td>3.77</td>
<td>2.33</td>
<td>0.86</td>
<td>0.34</td>
</tr>
<tr>
<td>40</td>
<td>0.77</td>
<td>2.00</td>
<td>3.12</td>
<td>1.75</td>
<td>0.80</td>
<td>0.33</td>
</tr>
<tr>
<td>80</td>
<td>0.56</td>
<td>1.56</td>
<td>2.19</td>
<td>1.38</td>
<td>0.73</td>
<td>0.31</td>
</tr>
<tr>
<td>120</td>
<td>0.52</td>
<td>1.17</td>
<td>1.53</td>
<td>1.05</td>
<td>0.62</td>
<td>0.30</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

Having the measured number of fissions and captures in the natural uranium foils we get (obtain) the spectral indexes (see Table 4).

Table 4. Fission to capture ratio – spectral indexes.

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/Z mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>1.649</td>
<td>2.297</td>
<td>0.953</td>
<td>0.651</td>
<td>0.412</td>
</tr>
<tr>
<td>40</td>
<td>0.688</td>
<td>1.180</td>
<td>1.176</td>
<td>0.943</td>
<td>0.637</td>
<td>0.424</td>
</tr>
<tr>
<td>80</td>
<td>0.446</td>
<td>0.646</td>
<td>0.735</td>
<td>0.594</td>
<td>0.425</td>
<td>0.323</td>
</tr>
<tr>
<td>120</td>
<td>0.346</td>
<td>0.521</td>
<td>0.542</td>
<td>0.467</td>
<td>0.355</td>
<td>0.300</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

- Having in turn the measured spectral index equal to ratio of average fission and average capture cross section we can evaluate the average neutron flux in the location of the actinide sample using the measured amount of fissioned and captured actinide isotopes.

- This is done by applying the try and error method where we look for the neutron energy for which the ratio of fission cross section to capture cross section of the selected actinide isotope from the nuclear data base is equal to the measured ratio of fissioned and captured actinide isotopes.

- Since the measured ratio is defined as the ratio of average fission and capture cross sections so the retrieved distinct fission and capture cross sections for the distinct neutron energy from the nuclear data base (ENDF/B-VII.1) describe the average values.
2. Experimental part.

2.3. Measurement.

2.3.2. Results

Table 5. Average neutron energy distribution [MeV].

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/Z mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>1.360</td>
<td>1.390</td>
<td>1.315</td>
<td>1.290</td>
<td>1.21</td>
</tr>
<tr>
<td>40</td>
<td>1.29</td>
<td>1.330</td>
<td>1.330</td>
<td>1.312</td>
<td>1.284</td>
<td>1.23</td>
</tr>
<tr>
<td>80</td>
<td>1.248</td>
<td>1.290</td>
<td>1.300</td>
<td>1.275</td>
<td>1.230</td>
<td>1.20</td>
</tr>
<tr>
<td>120</td>
<td>1.100</td>
<td>1.270</td>
<td>1.270</td>
<td>1.200</td>
<td>1.200</td>
<td>1.08</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

Fig. 2. Average neutron energy distribution versus target length for four different radiiuses.
2. Experimental part.

2.3. Measurement.

2.3.2. Results

Table 6. Average fission and capture cross section distribution for $^{238}\text{U}[\text{barn}]$. 

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>$R/Z$ mm</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>0</td>
<td>0</td>
<td>0</td>
<td>0.1755</td>
<td>0.0755</td>
<td>0.052</td>
<td>0.030</td>
</tr>
<tr>
<td></td>
<td>40</td>
<td>0.0524</td>
<td>0.0913</td>
<td>0.0913</td>
<td>0.071</td>
<td>0.048</td>
<td>0.031</td>
</tr>
<tr>
<td></td>
<td>80</td>
<td>0.0340</td>
<td>0.0524</td>
<td>0.0587</td>
<td>0.044</td>
<td>0.0312</td>
<td>0.029</td>
</tr>
<tr>
<td></td>
<td>120</td>
<td>0.031</td>
<td>0.0423</td>
<td>0.0423</td>
<td>0.343</td>
<td>0.030</td>
<td>0.027</td>
</tr>
</tbody>
</table>

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2. Experimental part.

2.3. Measurement.

2.3.2. Results

Table 7. Average fission and capture cross section distribution for $^{235}$U[barn].

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/Z mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>$\sigma_f$</td>
<td>0</td>
<td>1.223</td>
<td>1.226</td>
<td>1.217</td>
<td>1.214</td>
</tr>
<tr>
<td></td>
<td>$\sigma_c$</td>
<td>0</td>
<td>0.089</td>
<td>0.088</td>
<td>0.092</td>
<td>0.093</td>
</tr>
<tr>
<td>40</td>
<td>$\sigma_f$</td>
<td>1.214</td>
<td>1.218</td>
<td>1.218</td>
<td>1.216</td>
<td>1.213</td>
</tr>
<tr>
<td></td>
<td>$\sigma_c$</td>
<td>0.093</td>
<td>0.091</td>
<td>0.091</td>
<td>0.092</td>
<td>0.093</td>
</tr>
<tr>
<td>80</td>
<td>$\sigma_f$</td>
<td>1.209</td>
<td>1.214</td>
<td>1.215</td>
<td>1.212</td>
<td>1.206</td>
</tr>
<tr>
<td></td>
<td>$\sigma_c$</td>
<td>0.095</td>
<td>0.093</td>
<td>0.092</td>
<td>0.094</td>
<td>0.096</td>
</tr>
<tr>
<td>120</td>
<td>$\sigma_f$</td>
<td>1.191</td>
<td>1.211</td>
<td>1.212</td>
<td>1.209</td>
<td>1.203</td>
</tr>
<tr>
<td></td>
<td>$\sigma_c$</td>
<td>0.102</td>
<td>0.094</td>
<td>0.094</td>
<td>0.095</td>
<td>0.097</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

• The average fission and capture cross sections for $^{238}\text{U}[\text{barn}]$ and $^{235}\text{U}[\text{barn}]$ presented in Table 6 and Table 7 are corresponding to individual elements of Table 5 in which average neutron energies are given at the location of the natural uranium foils.

• The obtained values for average fission and capture cross sections let us to evaluate the neutron fluencies distribution and average neutron flux distribution in the sub-critical assembly Quinta by help of the equations (Eq. 4 and Eq. 5). These are collected in Table 8, (Fig. 3) and Table 9 (Fig. 4).

\[ N_{yfs} = N_{yf8} + N_{yf5} = V_p \bar{\phi} N_8 \bar{\sigma}_{f8} t + V_p \bar{\phi} N_5 \bar{\sigma}_{f5} t = V_p \bar{\phi} N_8 t (\bar{\sigma}_{f8} + \frac{N_5}{N_8} \bar{\sigma}_{f5}) \]  \hspace{1cm} (4)

\[ N_{yc8} = V_p \bar{\phi} N_8 \bar{\sigma}_{c8} t \]  \hspace{1cm} (5)

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2. Experimental part.
2.3. Measurement.
2.3.2. Results

Table 8. Average neutron flux distribution [x 10⁹ cm⁻² s⁻¹].

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/Z mm</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>5.024</td>
<td>7.677</td>
<td>4.327</td>
<td>1.505</td>
<td>0.591</td>
</tr>
<tr>
<td>40</td>
<td>1.415</td>
<td>3.871</td>
<td>5.988</td>
<td>3.377</td>
<td>1.467</td>
<td>0.575</td>
</tr>
<tr>
<td>80</td>
<td>0.955</td>
<td>2.909</td>
<td>3.896</td>
<td>2.538</td>
<td>1.268</td>
<td>0.422</td>
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<tr>
<td>120</td>
<td>0.752</td>
<td>1.951</td>
<td>2.654</td>
<td>1.859</td>
<td>0.839</td>
<td>0.412</td>
</tr>
</tbody>
</table>
2. Experimental part.

2.3. Measurement.

2.3.2. Results

Fig. 3. Average neutron flux distribution versus target length for four different radiuses.
2. Experimental part.
2.3. Measurement.

2.3.2. Results

Table 9. Neutron fluency distribution \([x \times 10^{13} \text{ cm}^{-2}]\).

<table>
<thead>
<tr>
<th>No. Foil plates</th>
<th>1</th>
<th>2</th>
<th>3</th>
<th>4</th>
<th>5</th>
<th>6</th>
</tr>
</thead>
<tbody>
<tr>
<td>(R/Z \text{ mm})</td>
<td>0</td>
<td>123</td>
<td>254</td>
<td>385</td>
<td>516</td>
<td>647</td>
</tr>
<tr>
<td>0</td>
<td>0</td>
<td>10.34</td>
<td>15.79</td>
<td>8.906</td>
<td>3.096</td>
<td>1.215</td>
</tr>
<tr>
<td>40</td>
<td>2.919</td>
<td>7.966</td>
<td>12.32</td>
<td>6.950</td>
<td>3.019</td>
<td>1.184</td>
</tr>
<tr>
<td>80</td>
<td>1.965</td>
<td>5.988</td>
<td>8.018</td>
<td>5.223</td>
<td>2.609</td>
<td>0.867</td>
</tr>
<tr>
<td>120</td>
<td>1.547</td>
<td>4.015</td>
<td>5.462</td>
<td>3.826</td>
<td>1.837</td>
<td>0.849</td>
</tr>
</tbody>
</table>
2. Experimental part.
2.3. Measurement.
2.3.2. Results

Distribution of 23 natural uranium samples in the whole volume of deeply sub-critical Quinta assembly let us to determine the volumetric distribution of average neutron flux of a specified average energy what in turn it let us to determine the optimal place in the assembly for incineration of the actinide.

So far, such measurements are not performed in the world.

Fig. 4. Neutron fluency distribution versus target length for four different radiiuses.
3. Comparison of current and earlier neutron fluencies evaluation.

- The experimental results presented in [1, 2] also concerned the estimation of neutron fluencies by actinide spectral index method with the difference that earlier Np-237 was applied [1]. The same Quinta target was irradiated with the same pulsed proton beam of 0.66 GeV energy extracted from the Phasotron accelerator with the difference that in the current evaluation applying the natural uranium (natU)[2]. The total number of protons of the irradiation was equal to \(8.5 \times 10^{14}\) during the time of irradiation equal to 20580 seconds (5h 43 min). In the earlier experiment [1] the total number of protons of the irradiation was equal to \(7.78 \times 10^{14}\) during the time of irradiation equal to 11890 seconds (5h 15 min). It means that the neutron fluencies in the current evaluation should be higher.

- In the earlier work [1] of evaluation we have processed the experimental data of irradiated Np-237 actinide samples and silicon detectors directly placed on sections 2 and 4 of the target of QUINTA assembly (see Fig. 5) without lead shield-reflector. So we can compare only two points of the current work \(U_{14}\) and \(U_{34}\) (see Table 1) which are 4 cm below the surface of the sections.
3. Comparison of current and earlier neutron fluencies evaluation.

- We have processed the experimental data of irradiated Np-237 actinide samples and silicon detectors directly placed on sections 2 and 4 of the QUINTA setup (see Fig. 5) without lead shield-reflector.
- These samples (Np-237) were 16 cm from the source of the neutron spallation source, what is about 4 cm further than the compared samples of natural uranium U$_{14}$ and U$_{34}$

Fig. 5 Target of QUINTA assembly.
3. Comparison of current and earlier neutron fluencies evaluation.

- We can expect that the evaluation of neutron fluencies can be close to each other. The neutron fluency at the position $U_{14}$ is equal to $4.015 \times 10^{13}$ cm$^{-2}$ while at its counterpart is equal to $1.88 \times 10^{13}$ cm$^{-2}$ and at the position $U_{34}$ is equal to $3.826 \times 10^{13}$ cm$^{-2}$ while at its counterpart is equal to $1.197 \times 10^{13}$ cm$^{-2}$ (see Table 10).

Table 10. Compilation of neutron fluencies measured by two methods for two experiments in terms of average neutron energy.

<table>
<thead>
<tr>
<th>Section number and position in the Quinta target</th>
<th>2 Top</th>
<th>4 Top</th>
<th>$U_{14}$</th>
<th>$U_{34}$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Neutron fluency[n/cm$^2$] x10$^{13}$</td>
<td>1.88</td>
<td>1.197</td>
<td>4.015</td>
<td>3.826</td>
</tr>
<tr>
<td>Neutron fluence using silicon detectors [n/cm$^2$] x10$^{13}$</td>
<td>1.59</td>
<td>0.686</td>
<td>-</td>
<td>-</td>
</tr>
</tbody>
</table>
3. Comparison of current and earlier neutron fluencies evaluation.

- Neutron damage in the silicon detector placed in certain neutron spectrum collects the damage from whole spectrum starting from neutron energy as a threshold equal to 170 KeV to very high energy about 50 MeV irrespectively of its energy (see Fig. 6 – red line). It is an additive process. So the reverse dark current in the silicon detector is proportional to the average damage done by the neutrons of actually existing neutron spectrum in this range.

![Fig. 6. Cross section of neutron inducing damage in the silicon detector (red curve) and cross sections of neutron inducing fission and capture on Np-237 minor actinide.](image)

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National Centre for Nuclear Research, Poland
3. Comparison of current and earlier neutron fluencies evaluation.

- Thus, we can further conclude that since the fluence of neutrons is comparable in both experiments (see Table 10), a significant part of the fluence of neutrons comes from the energy range of 0.9 - 2.0 MeV.
4. Conclusions

• Actinide samples and planar silicon detectors can be used as neutron fluency detectors especially in the high neutron energy range that is difficult to measure.

• Both the natural uranium and the neptunium 237 actinides can be applied as high energy neutrons fluencies detectors.

• Since the measurements of fluence of neutrons by the spectral index method using natural uranium and neptunium 237 and the silicon detector method are quantitatively comparable to some extent, we can conclude that a significant part of the measured fluence of neutrons comes from the neutron energy range 0.9 - 2.0 MeV.
4. Conclusions

- In the case of irradiation of Am-241 under the conditions described above, the spectral index should reach about 4, which also indicates that the incineration of this actinide will be effective.
- It is widely known that the average neutron energy during the process of fission is about 2 MeV but during the process of spallation is about 3 MeV.
- The quotient of cross sections for fission and capture for these neutron energies (1, 2 and 3 MeV) gives information on incineration of minor actinides. This is clearly seen in Table 11 where are collected the mentioned parameters for these neutron energies extracted from the data base ENDF/B-VII.1.
4. Conclusions

• The minor-actinide-bearing blanket (MABB) concept [5, 6] is based on irradiation of such blankets containing a significant quantity of minor actinides in a solid solution mixed with uranium oxide which is to be located in radial blankets on the periphery of the outer core in a sodium-cooled fast reactor (SFR).

• Inside in the structure of fuel the average neutron flux can be very close to the average energy of fission neutrons what ensure that the incineration efficiency is within the range of 8 – 27 times higher for Np-237 and 4 - 23 times higher for Am-241 than production of next actinides (see Table 9 – previous slide).
4. Discussion.

Table 11. The fission/absorption ratios in function of neutron energy for neptunium 237 and americium 241.

<table>
<thead>
<tr>
<th>Neutron energy [MeV]</th>
<th>Fission cross section $\sigma(n,f)$ [barn]</th>
<th>Capture cross section $\sigma(n,\gamma)$ [barn]</th>
<th>$\sigma(n,f)/\sigma(n,\gamma)$</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Np-237</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>1.4587</td>
<td>0.17277</td>
<td>8.43213</td>
</tr>
<tr>
<td>2.0</td>
<td>1.7001</td>
<td>0.06090</td>
<td>27.9664</td>
</tr>
<tr>
<td>3.0</td>
<td>1.6609</td>
<td>0.032674</td>
<td>50.8339</td>
</tr>
<tr>
<td></td>
<td>Am-241</td>
<td></td>
<td></td>
</tr>
<tr>
<td>1.0</td>
<td>1.2615</td>
<td>0.292707</td>
<td>4.30977</td>
</tr>
<tr>
<td>2.0</td>
<td>1.8498</td>
<td>0.07717</td>
<td>23.9701</td>
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<tr>
<td>3.0</td>
<td>1.85973</td>
<td>0.02145</td>
<td>86.6827</td>
</tr>
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</table>
5. References

• [1] M. Szuta, S. Kilim, E. Strugalska-Gola, M. Bielewicz, N.I. Zamyatin, A. I Shafronovskaya, S. Tyutyunnikov; Comparison of two fast neutron fluence measurement methods based on Np237 fission to capture ratio measurement (spectral index) and a reverse dark current measurement of planar silicon detector; XXIII International Baldin Seminar on High Energy Physics Problems – “Relativistic Nuclear Physics & Quantum Chromodynamics”; Russia, Dubna, September, 2016.; *Baldin ISHEPP XXIII*; EPJ Web of Conferences **138** 10006 (2017), DOI: 10.1051/epjconf/201713810006


• [3] M.Yu. Artiushenko, V.A. Voronko, K.V. Husak, M.G. Kadykov, Yu.T. Petrusenko, V.V. Sotnikov, D.A. Irzhevskyi, S.I. Tyutyunnikov, W.I. Furman, V.V. Chilap; Investigation of the spatial and energy distributions of neutrons in the massive uranium target irradiated by deuterons with energy of 1…8 GeV; *ISSN 1562-6016. BAHT. 2013. №6(88) 170 - 174

• [4] M.Yu. Artiushenko, A.A. Baldin, A.I. Berlev, V.V. Chilap, O. Dalkhajav, V.V. Sotnikov, S.I. Tyutyunnikov, V.A. Voronk1, A.A. Zhadan; Comparison of neutron-physical characteristics of uranium target of assembly „QUINTA” irradiated by relativistic deuterons and 12C nuclei; *ISSN 1562-6016. BAHT. 2016. №3(103)
5. References.


• Thank you for the attention.