Zakład Energetyki Jądrowej i Badań Środowiskowych UZ3

Minor actinides incineration efficiency in ADS

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Minor actinides incineration efficiency in ADS.

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1. Introduction

- Minor actinides (MA: Np, Am, Cm) are generated in nuclear fuel during irradiation in reactor.
- these elements significantly contribute in the radio-toxicity and heat generation of spent nuclear
- The key issue for reducing the MA radio-toxicity is the partitioning and transmutation of MA in fast neutron of accelerator driven system (ADS).
- So the problem is to determine the optimal place in the ADS for incineration of the actinides. This can be done by application of actinides as average neutron energy detectors.

1. Introduction
1.1. Fast neutron MA incineration method.

- 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:
- $$N_{yf} = V_p \, \bar{\phi} \, N \, \bar{\sigma}_f \, t \qquad (1)$$
- $$N_{yc} = V_p \, \bar{\phi} \, N \, \bar{\sigma}_c \, t \qquad (2)$$

Where

V_p - actinide sample volume [cm^3],

$\bar{\Phi}$ - average neutron flux in the place of actinide sample location [$n/cm^2 \cdot s$],

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, γ) respectively [barns],

t - irradiation time [s].

1. Introduction
1.1. Fast neutron MA incineration method.

- 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
1.1. Fast neutron MA incineration method.

- The fast neutron MA incineration measurement method consists in utilizing neutron irradiated actinide samples for estimating 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 (α_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 MA incineration method.

- 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.

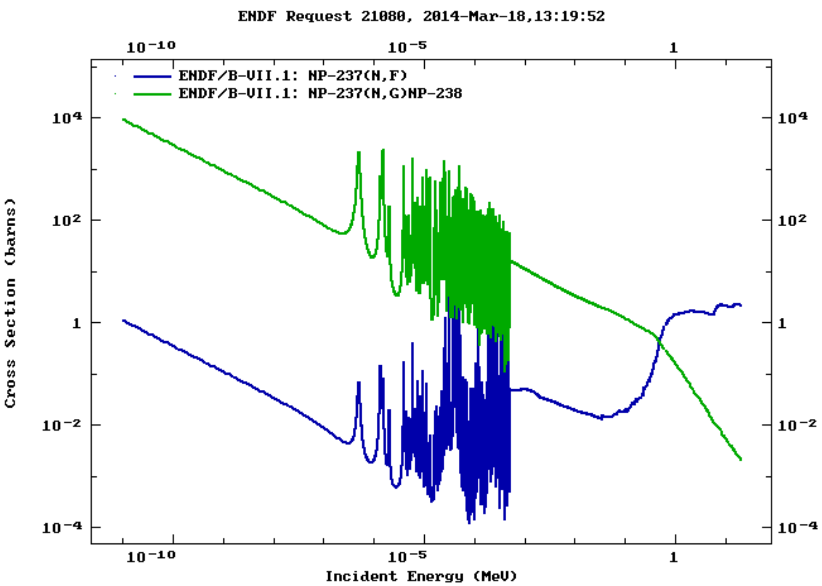

Fig. 1a. Cross-sections of Np-237(n,g)Np-238 and Np-237(n,f) reactions.

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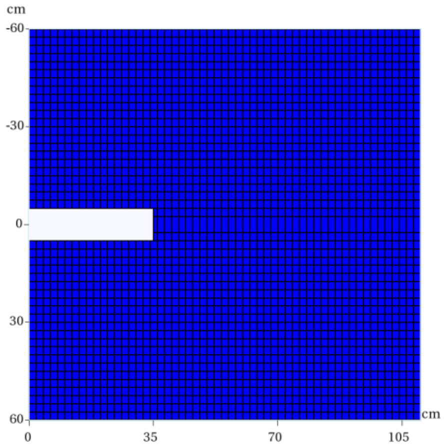
1. Introduction
1.1. Fast neutron MA incineration method.



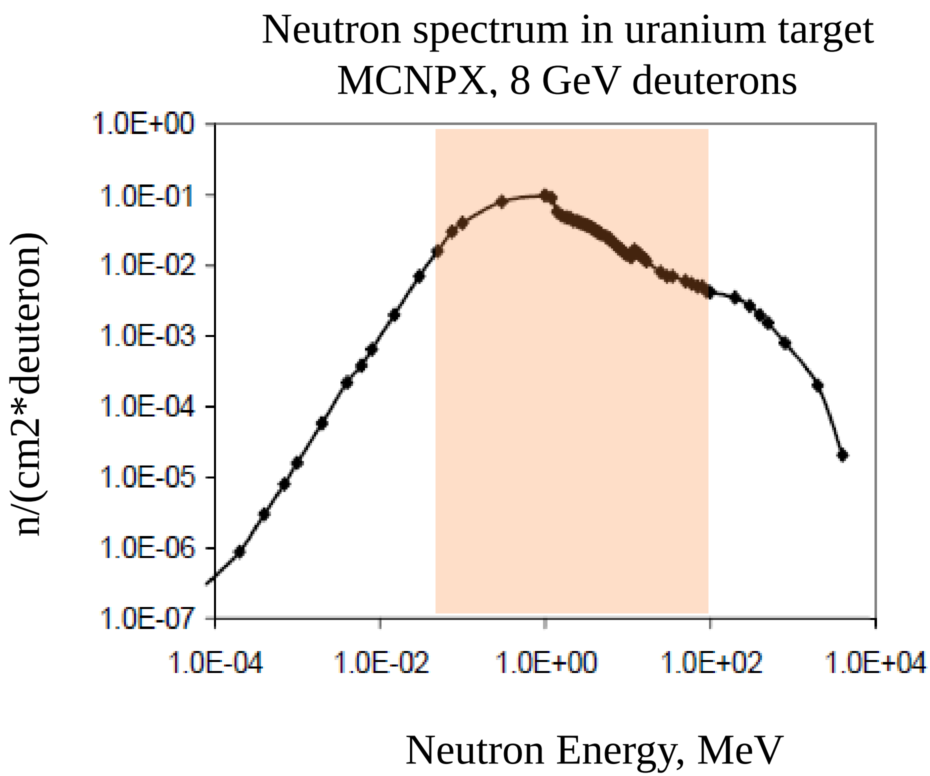
R = 60 cm, L= 110 cm, k_{eff} ~ 0.23

(*V.S. Pronskich et al. // Annals of Nuclear Energy. 2017, v. 109, p. 692-697*)

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

238U(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.



1. Introduction
1.1. Fast neutron MA incineration method.

Since the measured spectral indexes (α_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 ($\bar{\phi}$), as well the average fission and capture cross section values for the MA in the location of the actinide sample using the measured amount of fissioned and captured actinide isotopes.

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.
- $$^{238}\mathrm{U}(n,\gamma)^{239}\mathrm{U}\xrightarrow[23.5\,\mathrm{min}]{\beta}{}^{239}\mathrm{Np}\xrightarrow[2.36\,\mathrm{day}]{\beta}{}^{239}\mathrm{Pu} \qquad (3)$$
- (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:

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

$$N_{yc8} = V_p \bar{\phi} N_8 \bar{\sigma}_{c8} t \qquad (5)$$

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, γ) [barns],

$\bar{\Phi}$ - average neutron flux in the place of actinide sample location [$n/cm^2 \cdot s$],

$N_5 / N_8 = 0.7202/99.2752 = 0.00725 = 7.25\ 10^{-3}$

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 (α_{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.

$$\alpha_{m85}(E_d)=\frac{N_{yfs}}{N_{yc8}}=\frac{\bar{\sigma}_{f8}}{\bar{\sigma}_{c8}}+\frac{N_5}{N_8}\frac{\bar{\sigma}_{f5}}{\bar{\sigma}_{c8}} \qquad (6)$$

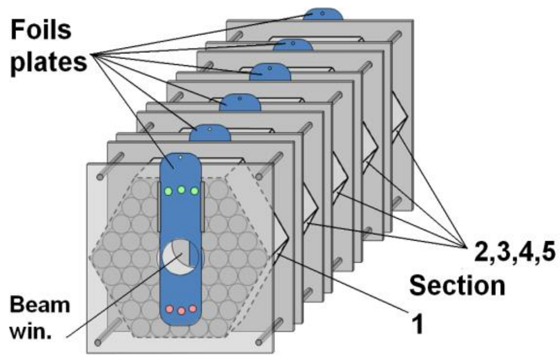
1. Introduction

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 $N_5/ N_8 = 0.7202/99.2752 = 7.25\ 10^{-3}$ 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:

$$\alpha_{m85}(E_d) = \frac{N_{yfs}}{N_{yc8}} = \frac{\bar{\sigma}_{f8}}{\bar{\sigma}_{c8}} + 7.25 10^{-3} \frac{\bar{\sigma}_{f5}}{\bar{\sigma}_{c8}} \qquad (7)$$

2. Experimental part.
2.1. Subcritical assembly Quinta



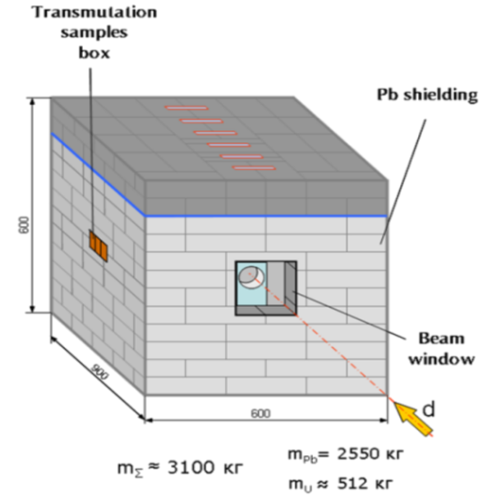
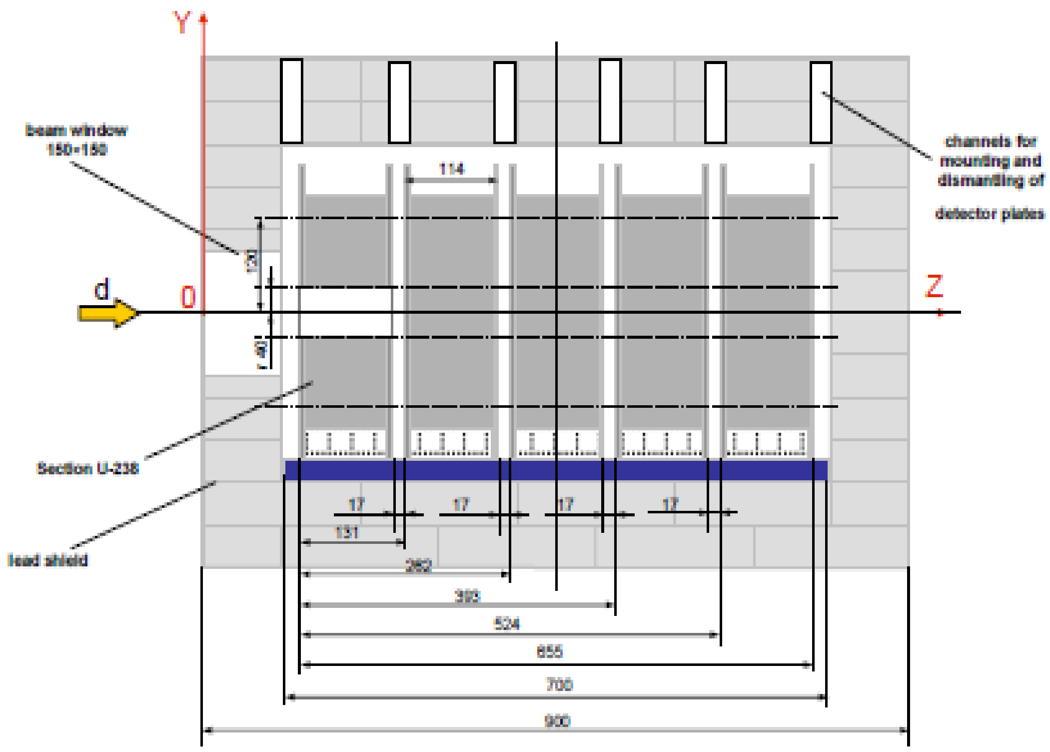


Fig. 1. Schema of Quinta assembly. On the left there is a view on the uranium target with supporting structures and plastics used for sample placement (detector's plates), on the right there is a view on the lead shielding enfolding the target with marked the transmutation samples box (window) for the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

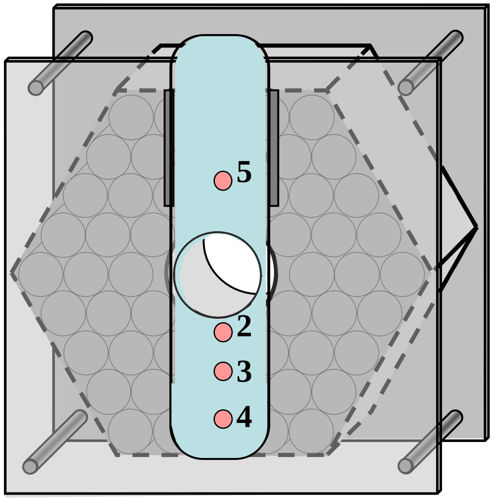
2. Experimental part.
2.1. Subcritical assembly Quinta

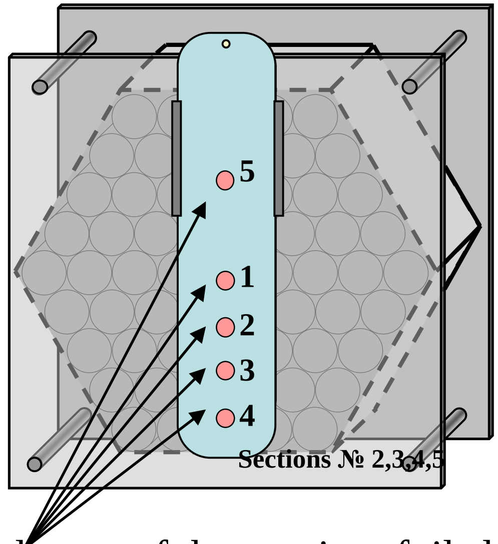
- Dimensions of Quinta assembly.

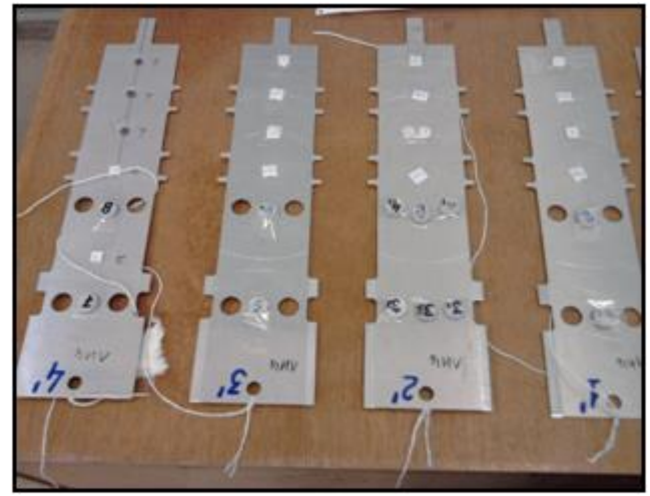


2. Experimental part.
2.1. Subcritical assembly Quinta

Sections № 2,3,4,5







Location of the detectors on the plate

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

- **5** R = -80 mm
- **1** R = 0
- **2** R = 40 mm
- **3** R = 80 mm
- **4** R = 120 mm

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.

R /Z, mm ***R - vertically*** ***Z - horizontally***	**Foil plates**					
	1	*2*	*3*	*4*	*5*	*6*
	0	123	254	385	516	647
0		U_{11}	U_{21}	U_{31}	U_{41}	U_{51}
40	U_{02}	U_{12}	U_{22}	U_{32}	U_{42}	U_{52}
80	U_{03}	U_{13}	U_{23}	U_{33}	U_{43}	U_{53}
120	U_{04}	U_{14}	U_{24}	U_{34}	U_{44}	U_{54}

2. Experimental part.
2.3. Measurement.
2.3.1. Irradiation details.

- The uranium target of assembly QUINTA was irradiated by 4.0 GeV deuterons at the accelerator Nuclotron JINR, Dubna.
- Monitoring of the deuteron beam was carried out by activation of aluminum and copper foils (see details in [6]).
- The total number of the primary deuterons hitting the target in this session of irradiation equal to 2.69 10^{13}, time of irradiation equal to 9 h.20 min (33600 s).
- Prior to the irradiation, several polaroid films were placed on the front of Quinta to ensure the deuteron beam was striking in the centre of the beam window.

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

- After the end of irradiation, the activation detectors taken out from the target to measure γ-spectra using HPGe detectors. Detection efficiency curves of detectors for various measurement positions were constructed using the following standard gamma sources: ^{54}Mn, 57Co, ^{60}Co, ^{88}Y, ^{109}Cd, ^{113}Sn, ^{133}Ba, ^{137}Cs, ^{139}Ce, ^{152}Eu, ^{228}Th and ^{241}Am.
- 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 reaction rates $Nx10^{-5}$, fission/g/deuteron

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	37.5	71.4	36.1	14.1	4.9
40	3.1	15.3	28.7	17.6	9.7	3.7
80	1.8	7.6	13.5	9.5	5.5	2.7
120	1.5	4.3	7.1	6.0	3.5	1.9

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, $Nx10^{-5}$, 239Pu/g/p

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	15.5	27.6	19.7	11.4	4.8
40	4.6	13.7	23.0	17.0	10.1	4.4
80	3.7	10.5	18.1	13.5	8.8	3.8
120	3.5	8.2	11.5	9.6	6.8	3.5

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.

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	2.419	2.598	1.832	1.237	1.021
40	0.674	1.117	1.248	1.035	0.960	0.841
80	0.486	0.724	0.746	0.704	0.625	0.711
120	0.429	0.524	0.617	0.625	0.515	0.543

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

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	1.396	1.403	1.369	1.335	1.319
40	1.288	1.328	1.335	1.320	1.315	1.306
80	1.253	1.294	1.297	1.292	1.282	1.293
120	1.232	1.261	1.280	1.281	1.259	1.265

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

Fig. 2. Average neutron energy distribution versus target length for four different radiuses.

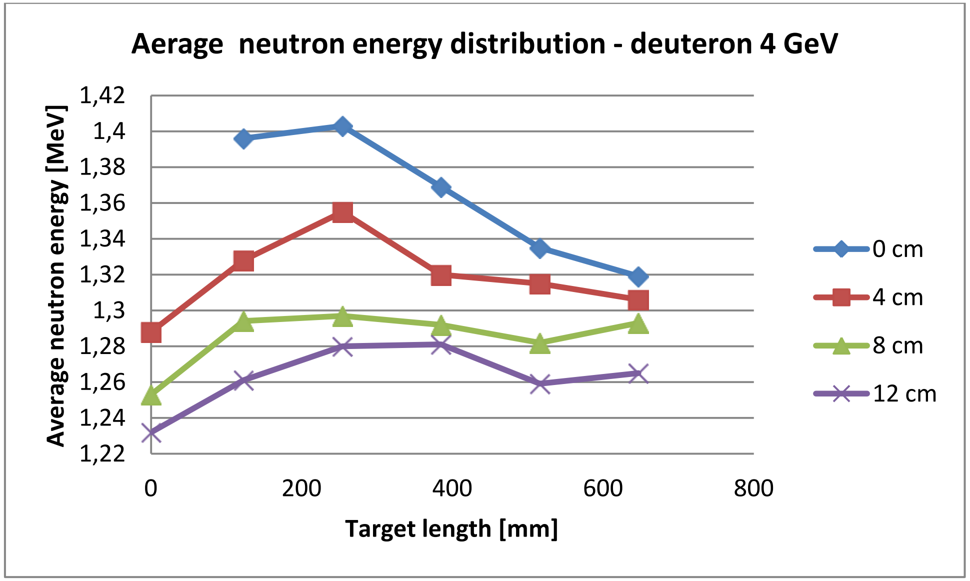

Fig. 2. Axial distribution of estimated average neutron energy for different radiuses.

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

Table 6. Average fission and capture cross section distribution for ^{238}U[barn] for 4 GeV.

No. Foil plates		*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm		*0*	*123*	*254*	*385*	*516*	*647*
0	σ_{f8}	0	0.1850	0.1982	0.1427	0.0967	0.0794
	σ_{c8}	0	0.0803	0.0797	0.0824	0.0851	0.0864
40	σ_{f8}	0.0511	0.0857	0.0967	0.0804	0.0750	0.0652
	σ_{c8}	0.0889	0.0891	0.0851	0.0863	0.0867	0.0874
80	σ_{f8}	0.0359	0.0549	0.0568	0.0537	0.0474	0.0543
	σ_{c8}	0.0917	0.0884	0.0882	0.0886	0.0894	0.0885
120	σ_{f8}	0.0315	0.0389	0.0461	0.0467	0.0381	0.0404
	σ_{c8}	0.0945	0.0910	0.0895	0.0894	0.0912	0.0907

2. Experimental part.

2.3. Measurement.

2.3.2. Results

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

No. Foil plates		*1*	*2*	*3*	4	*5*	*6*
R/Z mm		*0*	*123*	*254*	385	*516*	*647*
0	σ_{f5}	0	1.2271	1.2278	1.2237	1.2196	1.2118
	σ_{c5}	0	0.0882	0.0878	0.0895	0.0911	0.0919
40	σ_{f5}	1.2139	1.2187	1.2196	1.2178	1.2172	1.2161
	σ_{c5}	0.0935	0.0914	0.0911	0.0918	0.0921	0.0926
80	σ_{f5}	1.2097	1.2146	1.2150	1.2144	1.2132	1.2145
	σ_{c5}	0.0953	0.0932	0.0930	0.0933	0.0938	0.0932
120	σ_{f5}	1.2071	1.2106	1.2129	1.2131	1.2105	1.2111
	σ_{c5}	0.0963	0. 0949	0.0939	0.0939	0.0950	0.0947

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

- The average fission and capture cross sections for ^{238}U[barn] and ^{235}U[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).

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

$$N_{yc8} = V_p \bar{\phi} N_8 \bar{\sigma}_{c8} t \qquad (5)$$

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

Table 8. Average neutron flux distribution [x 10^8 cm^{-2} s^{-1}].

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	6.113	10.967	7.575	4.048	1.759
40	1.638	4.869	8.559	6.036	3.689	1.594
80	1.214	3.761	6.499	4.824	3.121	1.359
120	1.172	2.855	4.069	3.401	2.361	1.142

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

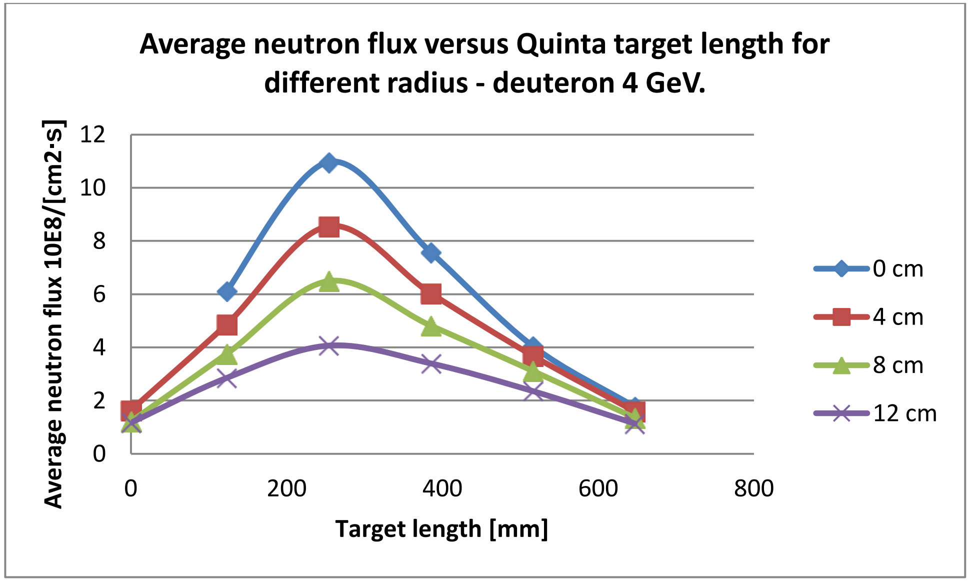


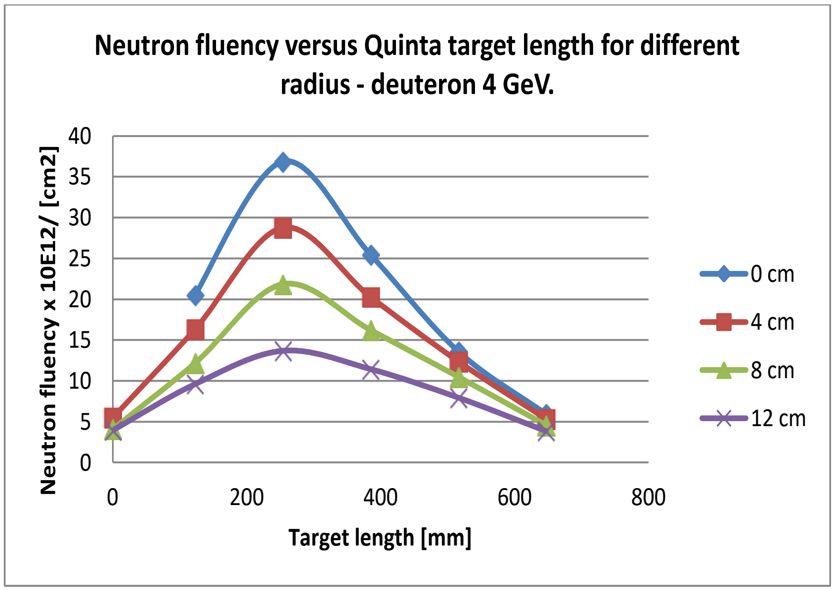
Fig. 3. Average neutron flux distribution versus target length for four different radiuses.

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

Table 9. Neutron fluency distribution [x 10^{12} cm^{-2}].

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	20.539	36.849	25.453	13.601	5.911
40	5.506	16.361	28.759	20.283	12.396	5.357
80	4.079	12.174	21.836	16.211	10.486	4.569
120	3.941	9.594	13.672	11.426	7.934	3.837

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 radiuses.

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.1 for the case of Np for the experiment of 4 GeV deuteron beam

- Having the average neutron energy distribution in the subcritical assembly Quinta (see Table 5 and Fig. 2):

- we can estimate for each location of the 23 locations:

- the average cross sections of fission and capture using the data base of Np, Am and Cm. (data base ENDF/B-VII.1.)

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.1 for the case of Np for the experiment of 4 GeV deuteron beam

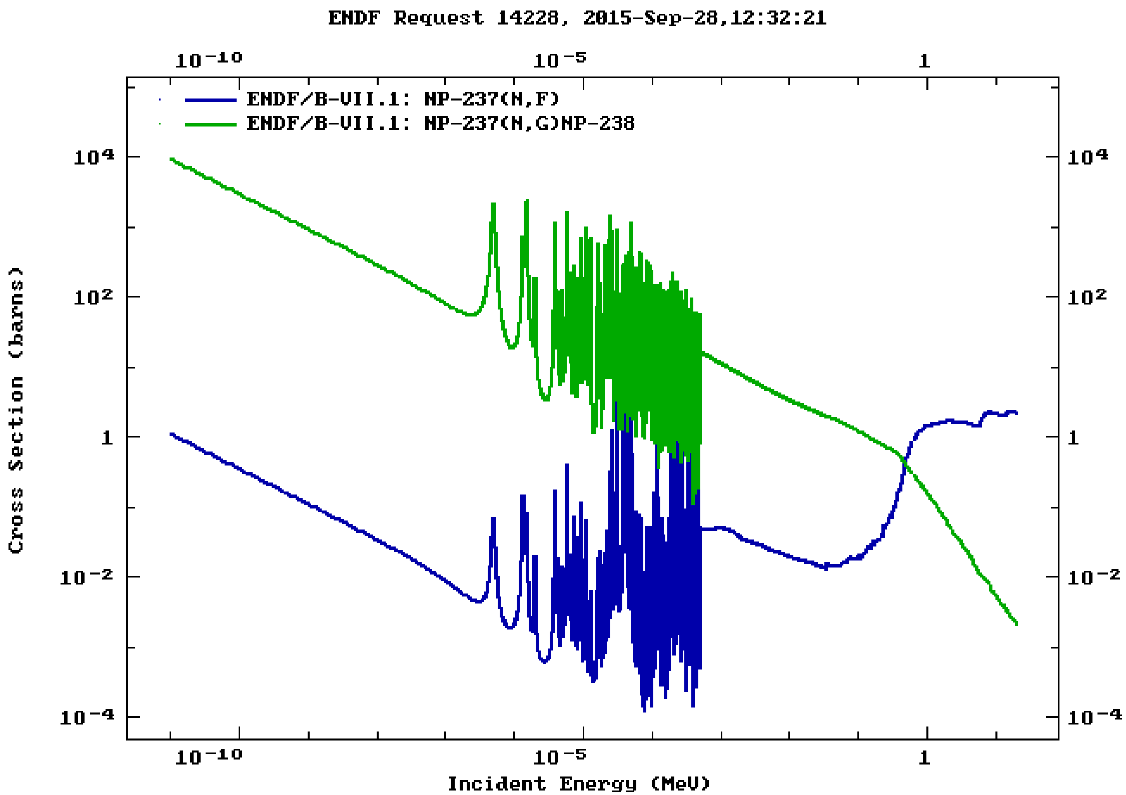


Fig. 1. Cross-sections of Np-237(n,g)Np-238 and Np-237(n,f) reactions.

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.1 for the case of Np for the experiment of 4 GeV deuteron beam

Table 10. Average fission and capture cross section distribution for Np-237[barn]; Experyment dec 2012, Edeut = 4 GeV, Quinta (ES-G)

No. Foil plates		*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm		*0*	*123*	*254*	*385*	*516*	*647*
0	σ_{f7}	0	1.5893	1.5916	1.5804	1.5692	1.5640
	σ_{c7}	0	0.1039	0.1030	0.1071	0.1103	0.1118
40	σ_{f7}	1.5564	1.5669	1.5692	1.5643	1.5626	1.5597
	σ_{c7}	0.1148	0.1110	0.1103	0.1117	0.1122	0.1131
80	σ_{f7}	1.5526	1.5571	1.5574	1.5568	1.5558	1.5569
	σ_{c7}	0.1181	0.1142	0.1139	0.1144	0.1154	0.1143
120	σ_{f7}	1.5399	1.5535	1.5555	1.5556	1.5532	1.5539
	σ_{c7}	0.1216	0.1173	0.1155	0.1154	0.1175	0.1170

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.1 for the case of Np for the experiment of 4 GeV deuteron beam

- Table 11. Fission/Capture DISTRIBUTION for Np-237 (spectral indexes)

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	15.2959	15.4482	14.7568	14.2237	13.9834
40	13.5593	14.1178	14.2237	13.9983	13.9244	13.7930
80	13.1460	13.6325	13.6694	13.6080	13.4867	13.6203
120	12.6604	13.2384	13.4627	13.4747	13.2152	13.2850

- These spectral indexes lie in the range 12.66-15.44 which proves that in such a neutron flux the incineration of this actinide Np-237is very effective.

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.2 Spectral indexes of minor actinides for different beam energies.

- Below are collected the ranges of spectral indexes of minor actinides (MA: **Np, Am and Cm**) for:
- three deuteron beam energy (2, 4 and 8 GeV),
- one for proton beam energy (0.66 GeV) and
- one for carbon beam energy (24 GeV).

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.2 Spectral indexes of minor actinides for different beam energies.

- The ranges of spectral index for Np-237
- 10.1536 - 15.1674 for 0,66 GeV proton beam
- 12.9333 - 15.2959 for 2 GeV deuteron beam
- 12.66 - 15.44 for 4 GeV deuteron beam
- 13.2733 - 15.9523 for 8 GeV deuteron beam
- 13.4867 - 14.7245 for 24 GeV carbon beam

3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.2 Spectral indexes of minor actinides for different beam energies.

- The ranges of spectral index for Am-241
- 5.6244 - 10.1587 - for 0,66 GeV proton beam
- 8.2667 - 10.7662 for 2 GeV deuteron beam
- 8.0733 - 10.8883 for 4 GeV deuteron beam
- 8.6142 - 11.3079 for 8 GeV deuteron beam
- 8.9166 - 10.2723 for 24 GeV carbon beam

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3. Estimation of spectral index for minor actinides Np, Am and Cm.

3.2 Spectral indexes of minor actinides for different beam energies.

- The ranges of spectral index for Cm-242
- 8.6535 - 10.0905 for 0,66 GeV proton beam
- 9.4260 - 10.1214 for 2 GeV deuteron beam
- 9.3836 - 10.1536 for 4 GeV deuteron beam
- 9.5073 - 10.2566 for 8 GeV deuteron beam
- 9.5581 - 9.9730 for 24 GeV carbon beam

4. Estimation of incineration rate for minor actinides Np, Am and Cm.

4.1 for the case of Np for the experiment of 4 GeV deuteron beam

- Having the distribution of the average neutron energy (Fig. 2, Table 5) and the average neutron flux density (Fig. 3, Table 8) along the length of the Quinta set for a deuteron beam with an energy of 4 GeV, we can first determine the distribution of cross-sections for fission and capture, e.g.Np-237 and then determine the distribution of the incineration rate of Neptune in the Quinta set. (Table 12)
- (data base ENDF/B-VII.1.)

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4. Estimation of incineration rate for minor actinides Np, Am and Cm.

4.1 for the case of Np for the experiment of 4 GeV deuteron beam

- Table 12._Np-237. Number of fissions/s for Np-237 = 1g [x 10^6 fissions/s].

No. Foil plates	*1*	*2*	*3*	*4*	*5*	*6*
R/Z mm	*0*	*123*	*254*	*385*	*516*	*647*
0	0	1.851	3.325	2.280	1.210	0.524
40	0.486	1.400	2.558	1.798	1.098	0.474
80	0.359	1.115	1.928	1.430	0.925	0.403
120	0.344	0.845	1.206	1.008	0.698	0.338

4. Estimation of incineration rate for minor actinides Np, Am and Cm.

4.1 for the case of Np for the experiment of 4 GeV deuteron beam

- There are two questions:
- -First question:

-The question is how many years we should irradiate the amount of 1g Np-237 in order to incinerate 0.5 g Np-237 (50%) in these conditions of Quinta subcritical assembly for the place $\mathbf{U_{21}}$? The answer is 2.416 10^7 years.

- Second question :

The question is how many years we should irradiate the amount of 1 g Np-237 in order to incinerate 0.5 g Np-237 in these conditions of Quinta subcritical assembly for the place $\mathbf{U_{21}}$ if the fission flux were equal to $\mathbf{5\ 10^{15}}$ **?**

The answer is **2.595 years.**

4. Estimation of incineration rate for minor actinides Np, Am and Cm.

4.1 for the case of Np for the experiment of 4 GeV deuteron beam.

- It means that in order to incinerate 0.5 g Np-237 (50%) in these conditions of Quinta subcritical assembly for the place $\mathbf{U_{02}}$ if the fission flux were equal to **5 $\mathbf{10^{15}}$ we have to irradiate about 2.6 years.**
- **So high neutron flux we can reach very easy in the spalation source.**

Shortly speaking.

- In order to present in short my speach I can mention the following steps:
- - measuremenr of fissions and cuptures of natural uranium,
- - evaluation of average neutron energy,
- - evaluation of average fission and cupture cross section,
- - evaluation of neutron flux distribution,
- - evalution of average fission and cupture cross section for analysed minor actinides,
- - evaluation of average fission rate considered minor actinide.

$$N_{yf} = V_p \, \bar{\phi} \, N \, \bar{\sigma}_f \, t$$

Shortly speaking.

$$N_{yf} = V_p \, \bar{\phi} \, N \, \bar{\sigma}_f \, t$$

$$N_{yc} = V_p \, \bar{\phi} \, N \, \bar{\sigma}_c \, t$$

5. Conclusions

- After embracing the experimental data and combining them we can suggest that the deuteron beam energy of 2 GeV controlling the subcritical assembly Quinta (ADS) in terms of efficiency incineration and economic view is the best one for incineration the MA.
- 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.

5. 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).

5. Conclusions

- It means that in order to incinerate 0.5 g Np-237 (50%) in these conditions of Quinta subcritical assembly for the place $\mathbf{U_{02}}$ if the fission flux were equal to **5 10^{15} we have to irradiate about 2.6 years.**
- **So high neutron flux we can reach very easy in the spalation source.**

5. Referencs

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- Thank you for the attention.

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