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MEASUREMENT OF THE NEUTRON COMPONENT IN A SHOWER GENERATED IN A LEAD TARGET BY RELATIVISTIC NUCLEAR BEAM

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1 Introduction

One of the main problems to be resolved under the JINR research project "Energy" is to find out optimal conditions for neutron generation in massive blocks of heavy metals (Pb, Bi, Th, U) by relativistic nuclear beams [1]. Methodic ensuring of this problem requires precise and reliable measurements of such values as total number of generated neutrons, beam fluence of the accelerator, energy spectrum of moderated neutrons, ratio of fission and radiation channels of slow neutron absorption in the target. These values being the primary ones determine in the result most important functions of optimization of physical and technical parameters such as mean energy worth of neutron generating, heat production, nuclear fuel breeding etc.

This paper dedicated to a method of determinating the total number of neutrons generated in a lead target.

2 Determination of the Total Number of Neutrons

2.1 Neutron Transfer in a Lead Target

The Lead target represents a rectangular parallelepiped with the dimensions of 50x50x80 cm^3. The beam of relativistic nuclei accelerated at the synchrophasotron with the cross "profile" of ~ 10 cm^2 reaches the central part of the target due to the enter channel with the depth of 20 cm.
A hadron shower of inelastic nuclear interactions products, particularly, nuclei, nucleons and pions is flamed up in the target. In the process of energy dissipation of the shower neutrons play special part. Neutrons as products of fragmentation, fission, spallation and complete destruction of the nuclei and in contrast to other charged products do not lose their energy for ionization and take part in nuclear reactions up to binding energy of the neutron in the Lead nucleus. The field of neutron generating in the reactions (n,2n) is completely kept within the target limits, consequently by the number of generating neutrons the target was equivalent to an infinite lead medium.

Such an affirmation is valid for Lead just because of its high fission threshold, but it is non correct for U and Th.

Subsequent moderation of neutrons takes place exclusively owing to elastic collisions with nuclei and the number of scattering before the emergence from the target in transversal direction to the beam is $3R^2/2\lambda^2$ and at $R = 25$ cm, $\lambda \simeq 3$ cm, it is no less than 100.

Natural Lead is sufficiently transparent for slow neutrons [2]: by absorption cross section ($\sigma_a = 0.17$ b) it is comparable to Zirconium ($\sigma_a = 0.1$ b). Thus a lead block of limited dimensions essentially represents a slow neutron source.

In order to localize neutron field in a small volume and simultaneously to enhance neutron registration sensitivity owing to increasing absorption cross section at low energy the neutron generating target is built up with a more strong than Lead moderator and absorber. Water was chosen as such a moderator. Macroscopic absorption cross section of water is equal to $\Sigma_a = 2.2\ m^{-1}$ and slowing down power is equal to $\xi\Sigma_s = 135\ m^{-1}$, where ξ is average logarithmic energy loss, Σ_s is macroscopic scattering cross section.

2.2 *Experimental setup*

Experimental setup of the target, beam monitors and neutron detectors is shown in fig. 1.

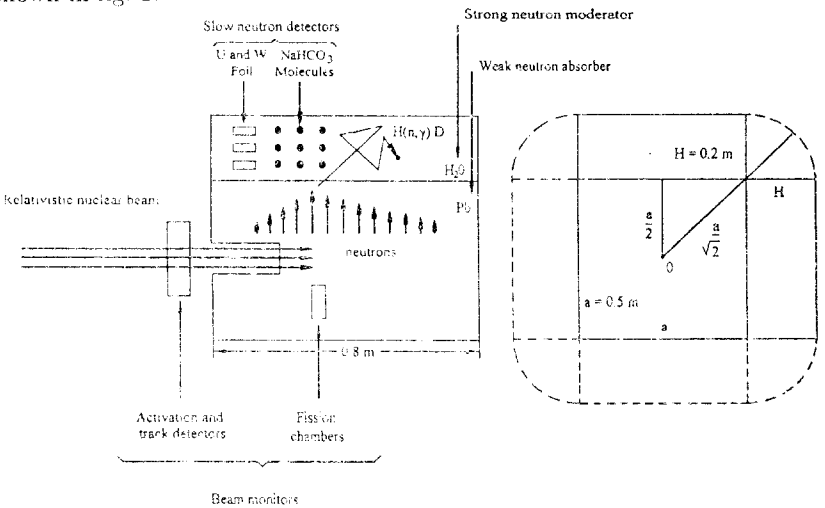

Fig. 1. Schematic diagram of the longitudinal and transversal sections of the setting

The beam axis coincides with the symmetry axis of the target. Parallel to the axis narrow slits were done for activation detectors from uranium foils in contact with solid state track detectors to be placed.

An organic glass bath filled with water was placed on the top of the target. Water represented simultaneously a neutron moderator, a medium for U and W detectors placing and a dissolvent of the soda detector.

Integral detectors for beam fluence monitoring are placed on the input of the target; fission chambers - monitors of secondary neutron intensity - are located inside of the target. Fission chambers are used for time factors evaluation thereby the number of nuclear interactions in activation detectors.

Here the time factor is dimensionless quantity: $T = \Sigma\, \omega_i \cdot T_i$,

where $\omega_i = \frac{\Phi_i \Delta t_i}{\Sigma \Phi_i \Delta t_i} = \frac{\Phi_i \Delta t_i}{\Phi_{tot}}$ - is weight of irradiation time,

$T_i = \frac{1}{\lambda \Phi_i} \left(1 - e^{-\lambda \Delta t_i}\right) \cdot e^{-\lambda \Delta t_{ai}} \left(1 - e^{-\lambda \Delta t_{mi}}\right)$,

Φ_i - secondary neutron flux (fission chamber reading).

This factor takes into account variation of the beam intensity during the exposition

2.3 *Procedure of extraction the Total Number of Neutrons from the Measurement data*

Most of the neutrons produced in the target are absorbed in water moderator because of transparency of natural Lead to slow neutrons. Therefore in the first the number of those neutrons has to be determined. For that reason using different activation detectors the number of nuclear interactions (n_n) within themselves is determined. Multiplying this number by the ratio of the moderator volume to the detector volume (V_m/V_d) and by the ratio of the macroscopic absorption cross section in Hydrogen to that in the detector (Σ_H/Σ_d) one can determine the number of neutrons absorbed in the moderator at $\Phi(r, E) = const$).

In order to determine the total number of neutrons it is necessary to take into account a number of corrections which can be conditionally divided into two groups. One of them is connected with neutron physics parameters both of the target and the moderator such as space distribution of the neutron flux, energy spectrum of slow neutrons and albedo on the boundary lead-target. Another group of corrections is connected with the construction features of the target and moderator, especially their forms and dimensions.

In section 4 each of these corrections will be considered. Here the summary correction is presented as Q, so the total number of neutrons generated in the target (N_n) is

$$N_n = n_n \cdot \frac{V_m}{V_d} \cdot \frac{\Sigma_H}{\Sigma_d} \cdot Q \qquad (1)$$

Alogarithm for determining the total number of generated neutrons described above is shown in fig. 2.

(i) Measured and evaluated information
- S_γ Area of photopeak in γ-rays spectrum
- ρ_{tr} Fission fragments track density
- n_n Number of nuclear interactions in detector
- N_n Total number of generated neutrons

(ii) Nuclear and atomic data
- $T_{1/2}$ Half-life of a nuclide
- E_γ Energy of γ-rays
- I_γ Quantum yield of γ-rays
- $\mu(E_\gamma, Z)$ Gamma-ray attenuation coefficient
- **R** Effective range of fission fragments

(iii) Parameters of the experiment
- ε Registration efficiency

Primary information
- S_γ
- ρ_v
Parameters of the experiment
- ε
- q_1, q_2, q_3
- $K(t)$
- G
- D
Number of nuclear interactions in detector
n_n
Total number of generated neutrons
N_n
Parameters depending on construction features
- V_m / V_D
- q_4, q_5, q_6
Nuclear and atomic data:
- $T_{1/2}$
- E_γ and I_γ
- $\mu(E_\gamma, Z)$
- χ
Parameters depending on material properties
- Σ_f / Σ_D
- q_1, q_2, q_3

Fig. 2. Scheme

t_1, t_2, t_3 Irradiation, cooling and measurement time

I(t) Beam intensity

G Measurement geometry factor

D Dimensions of detectors

(iv) Parameters depending on materials

Σ_H/Σ_d Ratio of macroscopic cross sections of neutron absorption in water and detector

q_1 Effect of space distribution of neutron

q_2 Effect of neutron spectrum

q_3 Effect of neutron absorption in target

(v) Parameters depending on construction

V_m/V_D Ratio of water and detector volumes

q_4, q_5, q_6 Correction factors related to the neutron escape through lateral sides, lateral edges and butt-end of the target, respectively.

2.4 Accuracy and Confidence of Determinating Neutrons Number

The reliability of optimazing the parameters of the supposed electro-nuclear reactor depends on the accuracy and the trustworthiness of determining the total number of neutrons generating in the massive target by accelerator beams. Therefore it makes high demands of this number determination technique. On the other hand this task is not only responsible but very difficult one because it is complicated to take into account a great number of parameters. Some of these parameters concerning to the nuclear spectroscopy and neutron physics are known with high precision, in the same time some of them require to be specified.

In order to improve the trustworthiness of neutrons number determinating a number of arbitrary parameters was minimized due to supplementary measuring all of the unknown values. Because of difficulties to describe processes of neutrons generating in intranuclear and internuclear cascades and in deexcitation of high energy nuclear states not a parameter of the model was used in the experiment, thus the measurement technique was model independent.

Because high statistical accuracy was reached in our experiments systematic errors were the main ones. There are principal difficulties in attempts to decrease those errors. Namely difficulties of precise calibrating of γ-rays and fission fragments registration efficiencies due to semiconducor and track detectors do not allow to reduce precision limits of our measurements less then $\pm 10\%$.

Criterion of confidence of the results was mutual consistence of essentially different parallel methods. In the next section four methods used for that reason are described in detail.

3 Determination of the Number of Nuclear Interactions in Detector Volume

3.1 Gamma Spectra of Activated Metallic Natural Uranium Foil

Choice of Uranium as an activation detector of slow neutrons do not connected with its advantages over other detectors, for instance in sensivity or precision. It is dictated by features of further investigation technique with an extended uranium target. Nevertheless Uranium as an activation detector posseses a number of merits:

- High abundance of the isotope ^{238}U (99.3%)
- High values of neutrons radiative capture cross section (σ_{th} = 2.68 b and I_r = 275 *b*)
- Reaction products have highly convenient half-lives for off-line experiments: ^{239}U (23.5 min) and ^{239}Np (2.35 days)
- Presence of sufficiently intensive lines in γ-rays spectrum of the daughter product ^{239}Np with energy 228 and 278 KeV which lies in the interval of the high efficiency of Germanium spectrometer.

This method was used successfully in [3,4] for investigation of neutron space distribution within Lead target. In the present paper side by side with other methods we use it for determinating the number of neutrons emerged from the target. In this case neutrons were moderated in water up to thermal energies. Metallic Uranium foils with thickness of 1 mm and area of 50 mm^2 were used as detectors. The weight of each of those detectors was approximately equal to 1 gramme.

The chosen thickness provided sufficient activation of the detector and in the same time coincided to the optimum self-absorption value of 228 and 278 KeV γ-rays. Self-absorption coefficient k = 0.63 was calculated for uniform activity distribution with the use of well known values of μ (see fig. 3) by formula:

$$k = \frac{1}{\mu d}(1 - e^{-\mu d})$$

where μ - is attenuation coefficient, d is thickness.

In the case of making neutrons space distribution in order to exclude influence of detectors weight differences the number of nuclei was normalized due to

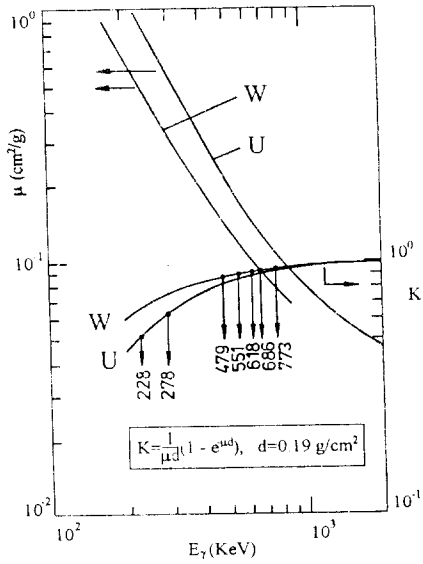

Fig. 3. Calculated values of $\gamma - rays$ self-absorption coefficients in U and W foils

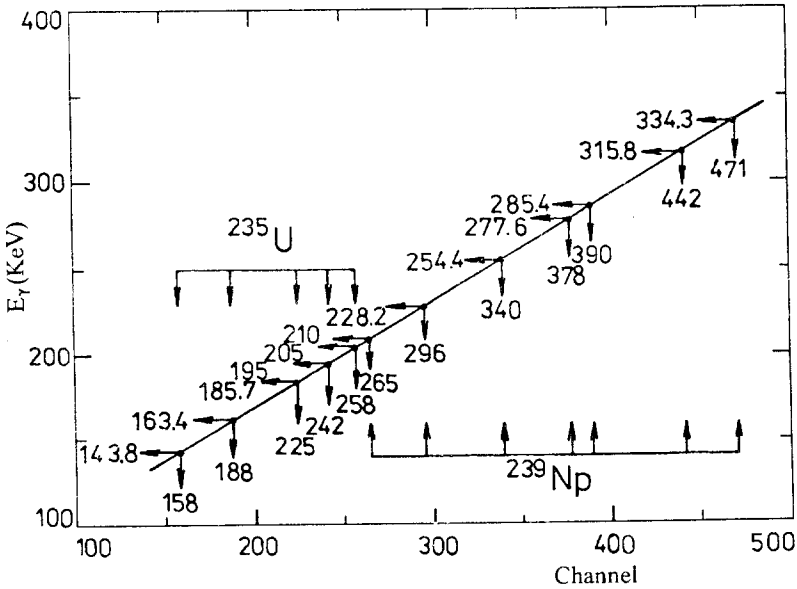

Fig. 4. Inner calibration of γ-rays energy of irradiated natural uranium foil

both weighting on precision balance and comparision of γ-rays intensity of the isotope ^{235}U for instance 186 KeV line.

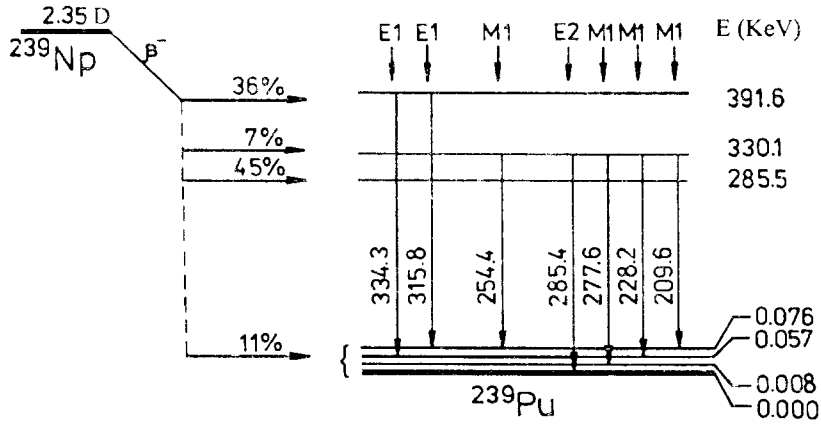

Fig. 5. Simplified scheme of β-decay of ^{239}Np

Energy calibration of the spectrometer was carried out with aid of not only external sources but also uranium detector itself, i.e. ^{235}U and ^{239}Np lines (see figs. 4 and 5). Calibration of spectrometer efficiency was carried out with the aid of high precise standard sources. Numerical coefficients a and b were found for each γ-spectrometer due to fitting the experimental data to the straight line in logarithmic co-ordinates and using them registration efficiency of γ-rays with the given energy was determined with an accuracy of 10%.

Gamma-rays of ^{239}Np with the energy of 277.62 KeV and quantum yield of $I = 0.145$ were the main information bearer of the number of neutrons absorbed in ^{238}U [5,6]. It is the most intensive line in the spectrum of this isotope, moreover, there are no other long-lived isotopes with the same energy among the products of the U(n,γ) and U(n,f) reactions. The spectrum of the isotope ^{134}Te which the fission product of ^{235}U contains 278 KeV line. But because of its short half-life (42 min) its influence can be neglected in spite of high yield (7%) if cooling time t_2, for all of irradiations is less than 2 hour ($t_2 \geq 3T_{1/2}$). In some cases when enhancing of statistics is required the second intensive line with the energy of 228.14 KeV and quantum yield of 0.114 was used. This line concretes with the 228.2 KeV, $I = 0.88$ line of the long-lived (78 h) isotope ^{132}Te which is the fission product of ^{235}U and is produced with the high yield (4%). Although these lines are difficult to be resolved at Ge(Li) spectrometer the presence of intensive lines such as 522 (15.6%), 630 (13.5%), 667.7 (98%), 772.7 (75%), 954.6 (16.7%) of daughter nucleus ^{132}I allows in principle to take into account contribution of ^{132}Te into the value of 228 KeV photopeak without spectral expansion. It is necessary to discuss other sources of possible

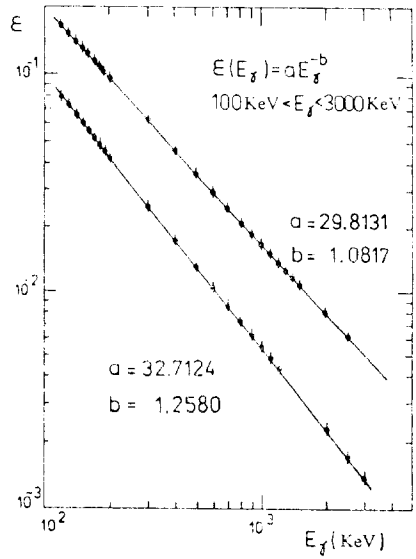

Fig. 6. Measured values of registration efficiency of γ-rays with the aid of Ge(Li)-detectors

errors. Natural radioactivity of Uranium-Radium and Thorium series contains γ-rays with energy close to two Neptunium lines: 279.4 (^{228}Ac), 277.35 (^{208}Tl), 226.1 (^{234}Pa), 228.5 (^{228}Th). However in our experimental conditions that background was negligible in comparision to Neptunium activity indicated in the Uranium detector.

When determining the number of nuclear interactions in the detector during irradiations it is necessary to take into account variations of the beam intensity. In section 2 monitoring of beam intensity with the aid of counter of secondary neutrons operating as multiscaler was discussed. Here we shall consider the time factor of ^{239}Np accumulation and decay during irradiation, cooling and measuring the uranium detectors. Proceeding from the comparision of half-lives of ^{239}U and ^{239}Np one can assume that on the expiry of cooling time t_{2_1}, which is equal to several ^{239}U half-lives, time factor is determined only by ^{239}Np decay as if it is a direct product of neutron capture in ^{238}U. Nevertheless we shall consider a general expression for T_1 and quantatively estimate the precision of such assumption.

From the Batheman-Rubinson equation [7,8] worked out for ^{239}U $\xrightarrow{\lambda_1}$ ^{239}Np $\xrightarrow{\lambda_2}$ ^{239}Pu chain it is easy to get:

$$T_i = \frac{1}{\lambda_2 t_{1i}}\left[\frac{\lambda_1}{\lambda_1-\lambda_2}(1-\mathrm{e}^{-\lambda_2 t_{1i}}) - \frac{\lambda_2}{\lambda_1-\lambda_2}(1-\mathrm{e}^{-\lambda_1 t_{1i}})\right]\mathrm{e}^{-\lambda_2 t_{2i}}(1-\mathrm{e}^{\lambda_2 t_3}) + \\ + \frac{1}{t_{1i}}\frac{\lambda_2}{\lambda_1-\lambda_2}(1-\mathrm{e}^{-\lambda_1 t_{1i}})\left[\frac{\mathrm{e}^{-\lambda_2 t_{2i}}}{\lambda_2}(1-\mathrm{e}^{-\lambda_2 t_3}) - \frac{\mathrm{e}^{-\lambda_1 t_{2i}}}{\lambda_1}(1-\mathrm{e}^{-\lambda_1 t_3})\right]$$

where the first term is related to Neptunium accumulated during the irradiation time t_{1i}, the second one is related to Neptunium produced after t_{1i} during the cooling time t_{2i}.

Opening the brackets gives

$$T_i = \frac{1}{t_{1i}}\left[\frac{\lambda_1}{\lambda_2(\lambda_1-\lambda_2)}(1-\mathrm{e}^{-\lambda_2 t_{1i}})\mathrm{e}^{-\lambda_2 t_{2i}}(1-\mathrm{e}^{-\lambda_2 t_3}) - \right. \\ \left. - \frac{\lambda_2}{\lambda_1(\lambda_1-\lambda_2)}(1-\mathrm{e}^{-\lambda_1 t_{1i}})\mathrm{e}^{-\lambda_1 t_{2i}}(1-\mathrm{e}^{-\lambda_1 t_3})\right]$$

Taking into account that $\lambda_2 \ll \lambda_1$ and neglecting the second item we obtain the following experession which is in agreement with above mentioned affirmation, i.e.

$$T_i \simeq \frac{1}{\lambda_2 t_{1i}}(1-\mathrm{e}^{\lambda_2 t_{1i}})\mathrm{e}^{-\lambda_2 t_{2i}}(1-\mathrm{e}^{\lambda_2 t_3})$$

The error which could appear after discarding the second term may be estimated from the ratio of the second and first terms:

$$\alpha = \frac{\lambda_2^2}{\lambda_1^2} \times \frac{1-\mathrm{e}^{-\lambda_1 t_{1i}}}{1-\mathrm{e}^{-\lambda_2 t_{1i}}} \times \frac{1-\mathrm{e}^{-\lambda_1 t_{2i}}}{1-\mathrm{e}^{-\lambda_2 t_{2i}}} \times \frac{1-\mathrm{e}^{-\lambda_1 t_3}}{1-\mathrm{e}^{-\lambda_2 t_3}}$$

The first multiplier is equal to $4.8 \cdot 10^5$, other three ones depend on a choice of times t_{1i}, t_{2i} and t_3 respectively, which are the parameters of the given experiment. Depending on the beam intensity of the accelerator (consequently on the activity which is induced in the detector) those times were chosen in our experiments approximately equal to $t_{1i} \simeq 0.2\ h$, $t_{2i} \simeq 10\ h$ and $t_3 \simeq 3\ h$, the values of three multipliers were equal to 120, 10^{-6} and 30, respectively, i.e. $\alpha < 10^{-6}$.

Because of small dimensions of the uranium foil ($d = 8$ mm) in comparision to Ge(Li) detectors volume (40 and 120 cm^3) used in the present experiment the value of the geometric factor G was assumed to be equal to 1 when the location of the Uranium foil concurs with center of the butt-end of germanium detector frame.

Knowing the values considered above functions of efficiency of the γ-spectrometer (ε_γ), self-absorption coefficient (K), quantum yield (I), geometric factor (G)

and time factor ($T_i = \Sigma\ \omega_i \cdot T_i$) and experimentally measured area of the 278 KeV photopeak (S_γ) the number of neutrons absorbed in uranium detector by radiative capture channel can be calculated. It is equal to the number of ^{239}Np nuclei produced at irradiating the target and is determined by formula:

$$n_n = \sigma_{th} \cdot N_N \cdot \Sigma\ \Phi_i \cdot t_{1i} = \frac{S_\gamma}{\varepsilon_\gamma \cdot K \cdot I \cdot G \cdot T} \tag{2}$$

3.2 Fission Fragments Registration Due to Solid State Track Detectors

Thick (20 μm) lavsan sheets were irradiated in contact with the same uranium foils which were activation detectors.

Lavsan detectors (polyethyleneterephtalate plastics) were etched in 6.25 N NaOH at 60° C for 90 min. At these conditions of treatment diameter of tracks of fission fragments reaches 2 - 2.5 μ which allows to scan easily under an optical microscope at the magnification 500X-600X.

The number of fission events, i.e. the number of neutrons absorbed by Uranium nuclei and caused their fission is determinded from registered track density ρ_{tr} (cm^{-2}) by formula:

$$n_n = \frac{\rho_{tr} \cdot S_d}{\varepsilon K i G} \tag{3}$$

where S_d - is the area of the uranium foil ($\rho_{tr} \cdot S_d$ - total number of tracks)
$\varepsilon = 0.90$ is registration efficiency
$K = 2 \times 10^{-3}$ is self absorbtion coefficient
$i = 2$ is the number of fission fragments per fission
$G = 1/2$ is a geometry factor

One can see that this formula is analogous to the formula obtained in the case of γ-rays measurements. However the corresponding numeral values differ from each other. For instance, because of large electric charge of fission fragments their registration efficiency is many times as high than in the case of γ-rays, but self-absorption coefficient is on the contrary many times as low.

The most serious problem for this technique is fission fragments absorption in the Uranium foil itself because fission fragments have a wide spectrum of mass, charge and energy. Therefore an average value is considered: effective range R_{eff}, and the ratio of effective range to the foil thichness is taken as self-absorption coefficient K.

In [9] the dependence of effective range of Uranium fission products on the

atomic numbers of the elements contained in the matter and the type of dielectric detectors which registered fission fragments was obtained. In the case of lavsan detectors effictive range of fission fragments in the Uranium was found to be equal to 5 mg $\cdot$ cm^{-2} with an accuracy of 10%. It means that the maximum depth from which fission fragments are escaped and registered in lavsan is equal to 2.67 μm. Hence for metallic uranium foil with the thickness of 10^3 μm self-absorption coefficient is equal to $k_\mu = 2.67 \cdot 10^{-3}$.

On the other hand the surface of the Uranium foil placed in air medium is gradually covered with the layer of Uranium oxide-protoxide U_3O_8 which can strongly decrease effective range of fission fragments. Using an empiric formula which is fit for practically any elemental composition of the medium.

$$R_{\text{eff}}\ (\text{mg} \cdot \text{cm}^{-2}) = 0.046 \sum \alpha_i \cdot Z_i + 0.78\ ,$$

where α_i is atomic concentration of i-th component

Z_i - its atomic number, we can be conviced that effective ranges in metallic Uranium (5 mg$\cdot$cm^{-2}) and in Uranium oxide-protoxide (2.2 mg$\cdot$cm^{-2}) differ from each other more than two times.

Thus determination of self-absorption coefficient of fission fragments in the uranium foil requires its surface to be cleaned up to metallic glitter. This procedure is sufficiently a difficult one, especially when a large number of foils is used. In this case we used the foils with uncleaned surface and self-absorption coefficient was determined due to the comparision of track densities from cleaned and uncleaned foils jointly irradiated in the neutron field. Track density from foils used by us accounts for 80% of the case of the clean metallic surface, i.e. $K = 0.8 \cdot K_\mu = 2.1 \cdot 10^{-3}$. This method of determinating the coefficient R is rightful only in that case when the thickness of the foils compared are identical and more than the thickness of the oxidation layer many times. Effective range corresponding to K found in this way is equal to 2.1 mg $\cdot$ cm^{-2} which is in coincidence with the value calculated above by the empiric formula for uranium oxide-protoxide. However this coincidence is only evidence of the fact that the thickness of the oxidation layer was no less than effective range.

3.3 Measurement of ^{24}Na activity in an Aqueous Solution of Soda

The choice of the isotope ^{23}Na as an activation detector of slow neutrons in our experiment was caused by a number of its properties:

- Activation cross sections by thermal and resonance neutrons 0.53 b and 0.31 b, respectively, are sufficient to achieve high sensitivity

- Half-life of the daughter nucleus ^{24}Na (15 h) is optimum for off-line measurements
- High quantum yield (> 95%) and high energy (1,37 and 2.75 MeV) of γ-rays allow to operate with a thick layer of weak solution.

Baking soda $NaHCO_3$ as a carrier of Na was chosen also because of its chemical and physical properties:

- Good solubility of soda in water in a wide range of concentration and temperature enables to prepare solution of high homogenezation. Baking soda in contrast to calcinated one does not produce a sediment with temperature variations.
- The other elements contained in soda molecule: H, C and O are not activated by slow neutrons and have low absorption cross sections of γ-rays. That is the advantage of soda over sodium chloride NaCl.
- Cheapness of soda and lack of aggressive effect are one of its advantages.

In water moderator with volume of 80 litres soda was dissolved up to concentration of 1-2%. Such a weak solution represented a distributed to whole bath "system of molecular detectors" of slow neutrons. The general experimental setup (fig. 1) was not changed in all the runs of measurements. However the volume and concentration of soda solution in the bath, activation detectors (U, W, U+SSNTD) setting in it and even the material of the bath were changed to optimize and calibrate different parameters of the experiment.
After each irradiation the solution in the bath is carefully intermingled for accumulated radioactive ^{24}Na nuclei to be homogenized. After that in order to determine the average number of nuclear interactions in the volume unit of the moderator a small part of the solution is separated ($0.25 \div 0.5$ l) and poured into a thin-walled (0.5 mg $\cdot$ cm^{-2}) cubic capacity.

The algorithm of determination of the number of nuclear interactions is the same as in the case of uranium foil which is in detail discussed in section 3.1.

γ-rays spectrum measurements were carried out at a Ge(Li) spectrometer of sufficiently large volume (120 cm^3) to increase registration efficiency of 1369 KeV γ-rays.

Product of geometric factor G and self absorption coefficient K for the solution sample was determined in the following way. Small quantity of soda previously activated by neutrons and measured at $G = 1$ was dissolved in the water of the same volume as sample's one. The solution obtained was poured into sample capacity and its activity was measured again at the same spectrometer at the same operating conditions. Due to comparision of the 1369 KeV photopeak areas in the spectra of dissolved and insolved soda with regard for the background and time shift the value of KG was found to be equal to 0.19. If one

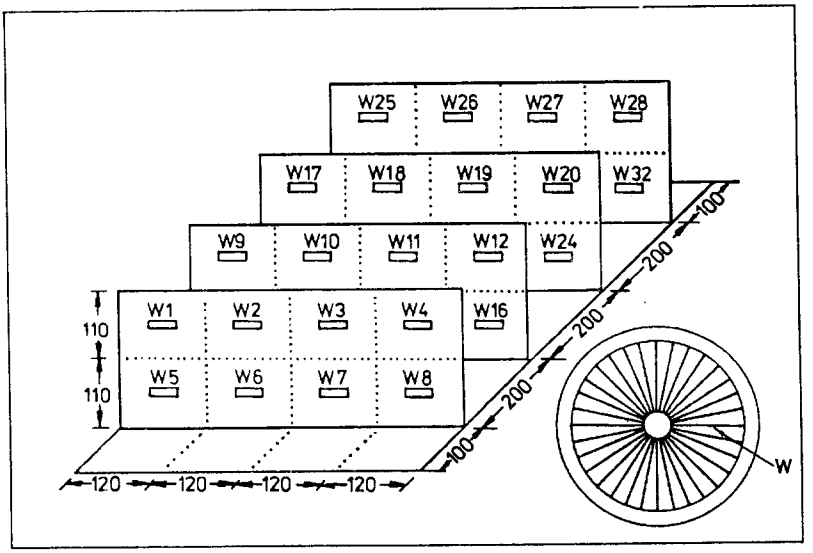

Fig. 7. Geometry of irradiation and measurement of Tungsten detectors

takes into account that energy of γ-rays is high and the average atomic charge Z of water is small, then $K = 1$ and $G = 0.2$.

3.4 Measurements of Tungsten Wire Detectors Activity

Each of methods considered in previous sections has its advantages: Uranium foils being concentrated in a small volume did not provoke a strong disturbance of the whole neutron field; soda solution as a detector distributed in a large volume did not require neutrons distribution to be measured. In order to combine those advantages into the one method parallel experiments with Tungsten detectors were carried out.

Tungsten was chosen as an activation detector for the following reasons. Considerable abundance of the ^{186}W (28.6%) isotope in conjunction with high slow neutrons absorption cross section of this isotope (40-400 b) ensure high sensitivity and accuracy. The facts that half-life of the daughter isotope ^{187}W is equal to 23.9 h and slow neutrons do not activate other isotopes of Tungsten essentially facilitate spectrometric measurements in off-line regime.

Tungsten wires with diameter of 1 mm and a length of 2 cm were used as neutron detectors. During the irradiation of the Lead target by accelerator beam 32 detectors were uniformly set in the moderator and after irradiation

their summary spectrum was measured at a γ-spectrometer in geometry of radial disposition (fig. 7). Geometric factor at such a location of detectors was found by special measurements to be equal to 0.7.

Determination of the average number of nuclear interactions in one tungsten detector was performed using several spectral lines: 479, 552, 618, 625, 686 and 773 KeV. These lines have sufficiently high intensity which is caused by a high population probabilities of ^{187}Re levels in the range of 600-800 KeV at β-decay of ^{187}W and a low multipolarities of electromagnetic transitions via which discharge of these levels takes place (figs. 8 and 9).

Morever we deal here with a highly rare decay scheme: all of these transitions are not cascade ones, i.e. the nucleus which emmit γ-quanta from one of these energies can not emmit γ-rays of other energies. This situation allows to write a formula for the number of nuclear interactions in a rather generalized form:

$$N_n = \frac{\sum_{i=1}^{m} S_{\gamma i} / \varepsilon_{\gamma i} K_i}{32\, G\, T \sum_{i=1}^{m} I_i} \qquad (4)$$

where i means summation over different spectrum lines. The value of m can vary from 1 to 6 at any combination of lines. However a combination ensuring sufficient statistical precision is prefered. At a high activity of detectors only one line is sufficient (479 or 686 KeV) and in this case the formula (4) gains the form of (2).

All of the values consisting in the formula (4) are determined in the same way as in the case of Uranium foils. Particularly, self-absorption coefficients K_i are determined from the known experimental data μ_i when an uniform activation of detectors is assumed. On the other hand such a assumption is not obvious because high absorption cross sections of slow neutrons in Tungsten may lead to the self- screening. Because of it values of K_i were verified by additional experiments which showed an absence of such an effect at the wire radius of 0.5 mm.

Finally, let us discuss one more detail of tungsten detectors technique. Because neutron absorption resonance integral in ^{186}W exceeds the corresponding thermal cross section by more than an order some parts of the moderator with low cadmium ratio in principle can provoke a distortion of the results.

In spite of the fact that the lower layer of the Tungsten detectors was located at a height of 5.5 cm from the bath bottom (fig. 7) and most of neutrons in the water layer of thickness less than 3 cm were in the epithermal range of energy (fig. 12) the latter did not exert influence upon the number of absorption in the detector averaged cover the whole bath volume.

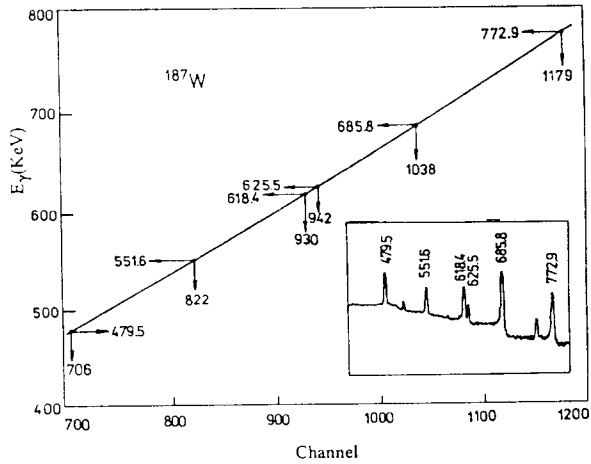

Fig. 8. Typical spectrum of ^{187}W γ-rays

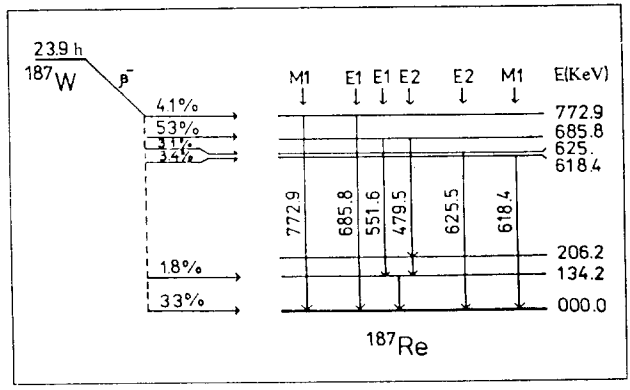

Fig. 9. β-decay scheme of ^{187}W

4 Determination of Parameters Depending on Materials and Construction

4.1 Space Distribution of Neutron Flux in the Moderator

Distribution of neutron flux in the target and the moderator was determined due to measuring the number of Uranium fission fragments by means of solid state nuclear track detectors. These distributions are shown in fig. 10, where

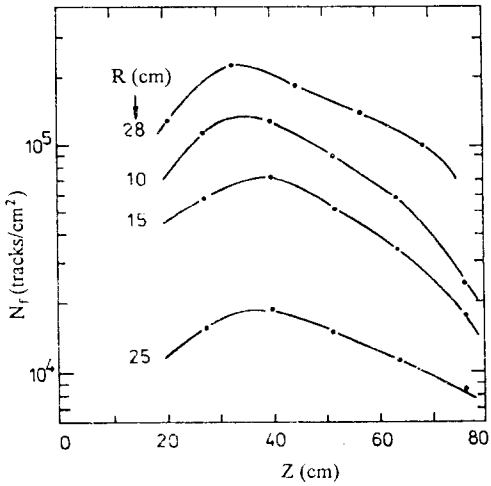

Fig. 10. Track density distribution inside of the target and in the moderator

Z is the distance from the front edge of the target, R is the distance from the beam axis. Distributions bearing onto the inner part of the target are in good agreement with the results of [3] where such distributions were measured by means of fission chambers.

Decreasing of the track density with mooving off to the beam axis is connected with geometry and its linear reduction is evidence of weak absorption of slowing-down neutrons in the target. Sharp increasing of the track density (10 times) at the transition from the target border (R = 25 cm) into the moderator (R = 28 cm) is caused by the enhancing of the fission cross section at low energies. The thickness of 3 cm of hydrogenous medium on this way (0.5 cm of organic glass and 2.5 cm of water layer) exactly coincides with the moderation length of neutrons in such substances.

Z-distribution of the track density in the moderator, which can be wittingly attributed to the ^{235}U fission by thermal neutrons, strongly resembles the distribution within the target where ^{238}U fission by fast neutrons has also to be of importance (fig. 10). Fig. 11 shows Z-distribution of neutron flux at R = 28 cm measured by means of two parallel methods: measuring the number of nuclear interactions due to radiation in ^{238}U and fission in ^{235}U. Those measurements were carried out with the same foils of natural Uranium. The coincidence of these two measurements is quite obvious and it is evidence of the correctness of the measured distributions. Data reffereing to the point of Z = 33 cm were used for determination of the number of nuclear interactions in the Uranium foil because it is the maximum point of the distribution and

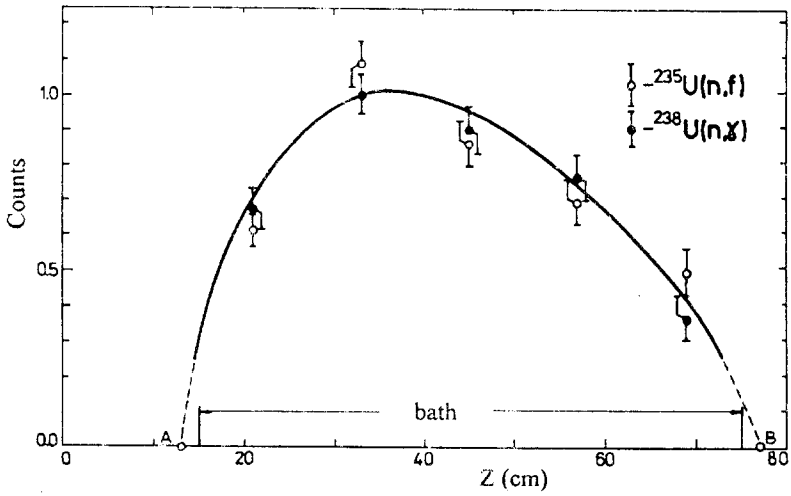

Fig. 11. Z-distribution of the neutron flux in the bath according to the data obtained by two parallel methods

due to the integration of the values normalized to this point we obtained the average of the neutron flux along the lenght of the bath. It was equal to 0.7. Analogical measurements showed that the average of the neutron flux along the width of the bath is also equal to 0.7 of the maximum in the half of the width.

Track density distribution along the bath height is shown in fig. 12 (white points). One can see from it that neutron field in fact utterly localized in the water layer of the thickness of 20 cm. The distribution achieves its maximum at the height of 3 cm. This fact testifies that at the first three centimeters not all the neutrons are moderated up to the thermal velocities. It is confirmed by measurements by means of Tungsten detectors high sensitive to epithermal neutrons (black points). The average flux density along the height obtained by means of integration of its normalized values was found to be equal to 0.4 of the maximum.

The parameter q_1 (see section 2.3) defining the variation of neutron flux spatial distribution can be expressed as a product of the average neutron flux in three reciprocally perpendicular directions, i.e. $q_1 = 0.7 \cdot 0.7 \cdot 0.4 = 0.2$. An absence of necessity to measure these parameters indicates the one of the advantages of the method of soda bath in comparision to the methods with localized uranium detectors.

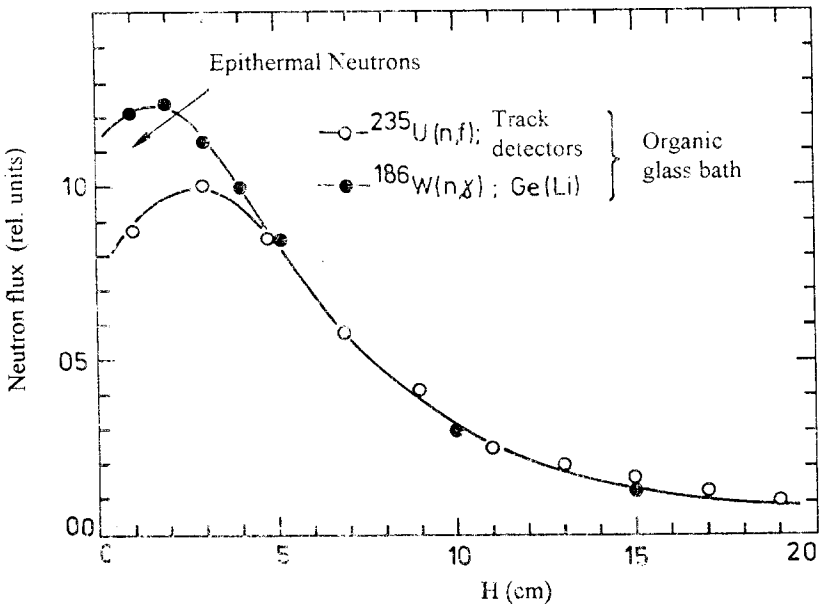

Fig. 12. Neutron flux distribution along the bath height

4.2 Energy spectrum of slow neutrons

It is easy to show that spectrum-averaged cross section σ_{eff} of any nuclear reaction in the field of slow neutrons, corresponding thermal cross section σ_{th} and resonance integral I are connected with each other by the following expression:

$$\frac{1}{\sigma_{\text{th}}} - \frac{1}{\sigma_{\text{eff}}} = \frac{1}{CR}\left[\frac{1}{\sigma_{\text{th}}} - \frac{1}{I_{\text{r}}}\right]$$

where CR - cadmium ratio. Fig. 13 shows calculated by means of this formula effective cross sections of the reactions used in the present investigations in the range of cadmium ratio from 1 to 1000.

In the case of $CR > 10$ effective cross sections are close to thermal cross sections and in the range of $1 < CR < 10$ their value can vary between σ_{th} and I_{r}. The cross section of the reactions ^{186}W(n,γ) and ^{238}U(n,γ) vary by 1 and 2 orders, respectively. Such strong variations are connected with a great number of resonances of these nuclei. This property of Tungsten was used for determination of an average cadmium ratio which is the integral characteristic of the spectrum.

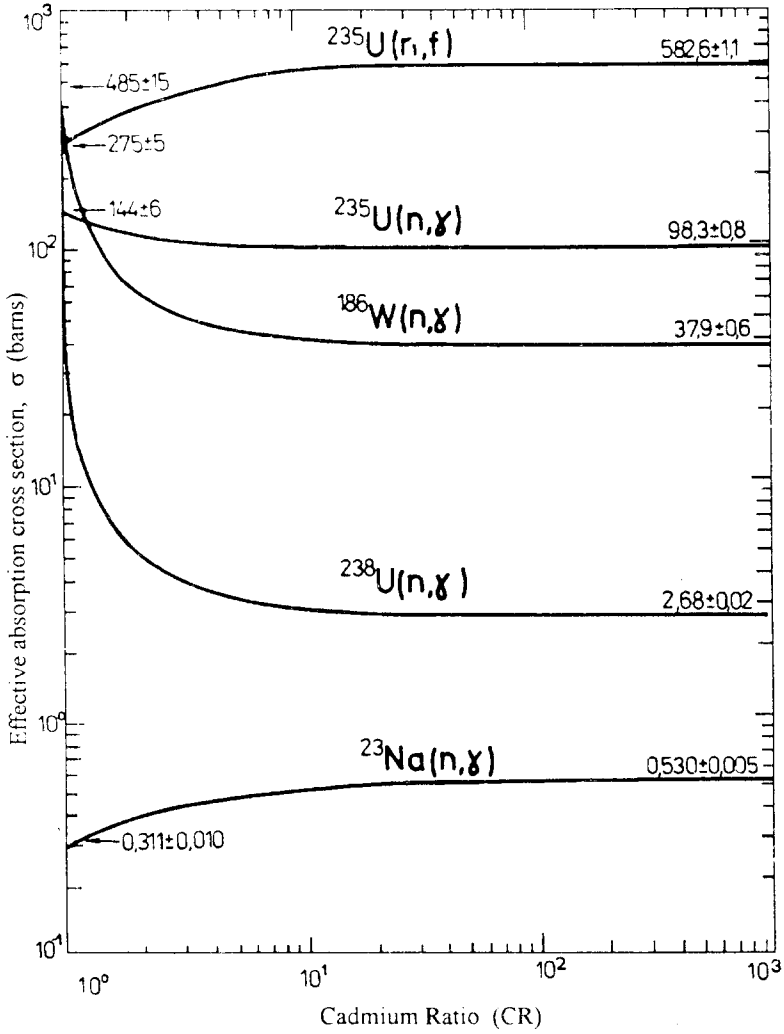

Fig. 13. Dependence of neutron absorption cross sections on cadmium ratio

In the last point we mentioned that epithermal neutrons were found in the water layer on the bath bottom by means of Tungsten detectors. One can see from the fig. 13 that the cross section of the $^{235}U(n,f)$ reaction decreases at the small CR values and the cross sections of the $^{186}W(n,\gamma)$ sharply increases. Just that very situation is reflected on the distributions measured by two different detectors and displayed as furcation in the fig. 12.

Nevertheless the reading of two detectors close in with penetrating to the depth of the moderator and merge in the end. This fact allows to assume that the resonance neutrons quota in the neutron spectrum integerated all over the moderator volume is small. The thrustness of this assumption was tested due to the comparision of the average number of ^{187}W nuclei produced in the one of the 32 tungsten detectors (section 3.4) with the number of ^{24}Na nuclei produced in the volume of soda solution sample during the same irradiation:

$$\frac{n_{\mathrm{W187}}}{n_{\mathrm{Na24}}} = \frac{N_{\mathrm{W186}}}{N_{Na23}} \times \frac{\sigma^{\mathrm{eff}}_{\mathrm{W186}}}{\sigma^{\mathrm{eff}}_{\mathrm{Na23}}}$$

The results of measurements of the number of nuclear interactions in the volume of two different detectors of definite type (see table 1) allowed to determine the ratio of "effective" cross sections of the ^{186}W(n,γ) and ^{23}Na(n,γ) reactions:

$$\frac{\sigma^{\mathrm{eff}}_{W186}}{\sigma^{\mathrm{eff}}_{\mathrm{Na23}}} = \frac{n_{\mathrm{W187}}/N_{\mathrm{W186}}}{n_{\mathrm{Na24}}/N_{\mathrm{Na23}}} = 69.5$$

The value obtained coincides with the ratio of thermal cross sections ($\sigma^{\mathrm{th}}_{W186}$: $\sigma^{\mathrm{th}}_{\mathrm{Na23}} = 71.3$) with an high accuracy. This fact allows to affirm that the average cadmium ratio of the neutron field is more than 50 ($\overline{CR} > 50$), i.e. the admixture of epithermal neutrons possible is less than 2%. Consequently there is no need to take into account deflection of the spectrum from the equilibrium, i.e. one can consider q_2 to be equal approximately to 1.

4.3 Neutron Albedo and Probability of Neutron Absorption in the Target

Absorption cross section of neutrons by the ^{207}Pb nucleus with one hole on the closed neutron shell ($N = 126$) exceeds absorption cross section by ^{208}Pb nucleus with closed neutron shell for 1400 times. Because of it macroscopic absorption cross section in natural lead remains fully appreciable and is equal to 0.0056 cm^{-1}. This value is only 4 times less more than macroscopic cross section in water which is equal to 0.0222 cm^{-1}. Therefore the definition of lead as weak absorber is highly relative conception and at the calculation of the total number of neutrons it is necessary to take into account the quota of neutrons absorbed in the target.

In section 4.1 it have been shown that the water layer of thickness of 3 cm on the bath bottom becomes an isotropic source of slow neutrons. It is naturally

to assume that some of these neutrons are back to the target and after numerous reflections and crossing the boundary between water and lead they are absorbed in these media. One can estimate the probability of absorption in Lead by means of the macroscopic absorption cross sections and the diffusion lengths of both media in the following way:

$$\beta = W_{Pb}/W_{H_2O} = \Sigma_{Pb} \cdot L_{Pb}/\Sigma_{H_2O} \cdot L_{H_2O},$$

where the ratio of diffusion length gives the ratio of times of neutron stay in both media. Using known experimental data[11] $\Sigma_{Pb} = 0.0056$ cm^{-1}, $\Sigma_{H_2O} = 0.022$ cm^{-1}, $L_{Pb} = 12.9$ cm, $L_{H_2O} = 2.77$ cm we obtain: $\beta = W_{Pb}/(1 - W_{Pb})$, i.e. $W_{Pb} = 0.54$

In order to measure separtely incident and reflected components of the thermal neutron scalar field we used "double sandwich" method, i.e. a pair of lavsan detectors in contact with uranium foil and defended from the other sides with cadmium screen. Those detectors were placed on the bath bottom near to each other for their readings could be refered to the same point of measurement. Over-cadmium background was measured by means of the third detector defended from two sides. Neutron source ^{252}Cf was situated over those detectors at the height of 3 cm. It's soft spectrum allowed to imitate the zone of neutron moderation which occured in the experiment with the beam.

Analogical measurements were simultaneously carried out with fission chambers KNT-8. The same result was obtained in both measurements for W_{Pb}, it was equal to 0.59 ± 0.01. This result is in satisfactory agreement with above considered estimate. Thus the parameter taking into account neutron absorption in lead is equal to $q_3 = 1/(1 - W_{Pb}) = 2.44$.

4.4 Transition to the 4π-geometry with Regard for Target Construction

Calculation of the total leakage of neutrons from the rectangular target using the results of measurements on the one of its sides proved to be an complicated task.

An assumption of additivity of four geometrically identic baths is rightful only in the case of simultaneous exposition and sufficient large volume.

In our experiment minimum distance between the points of the opposite sides ($a = 0.5$ m) was equal to four diffusion length of neutrons in lead so that an escape through them can be considered as additive one. In regard to the neighbour adjacent sides for which minimal distance is equal to 0 additivity is less obvious.

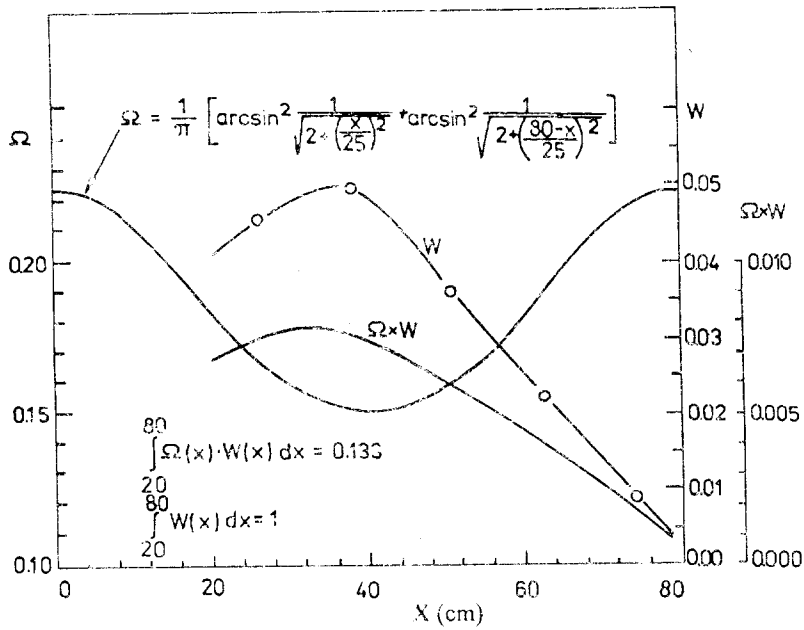

Fig. 14. Distribution of neutron leakage through butt-end sides of the target as a function of the co-ordinates of the sources

Rather large dimensions of the target ($a \gg L$) and axis symmetry of physical processes forming the neutron field in it allow to divide neutron escape into two parts:

- Leakage through the lateral sides and fields bordered upon the lateral sides (fig. 1)
- Leakage through the butt-end sides (fig. 14)

The minimum volume of the moderator on the lateral sides and within the dihedral angles at the lateral sides of the parallelepiped (fig. 1) is equal to $V = abH + \pi bH^2$, where H is the height of the water moderator and equal to the neutron total absorption length, a and b are the width and the length, respectively. Because the bath volume $V_0 = abH$, so $V = (1 + \pi H/4a) \cdot 4V_0 = 1.31 \cdot 4V_0 = 5.24V_0$, i.e. the volume of the moderator increases owing to the lateral dihedral angles by 31% in comparision with quadruplicated bath volume. An estimate for neutron spatial distribution in the conjugated baths is given by

$$q_1 V_0 + q_1^* 0.31 V_0 = q_1 V_0 (1 + \frac{q_1^*}{q_1} \cdot 0.31)$$

where q_1 and q_1^* are absorption coefficients of neutrons (see section 4.1).

Because of the axis symmetry the ratio q_1^*/q_1 has to be equal to the ratio of corrections of the flux variation along the moderator width correspondingly within the dihedral angle and in the bath. Direct measurements carried out by means of tungsten detectors showed that neutron flux linearly decreases along the width of the single bath from 1 (in the centre) up to 0.4 (on the lateral side of the bath), i.e. correction of neutron absorption along the bath width was equal to 0.7 (see section 4.1). Such a correction within the dihedral angle must be equal to the flux doubled value on the lateral side of the bath, i.e. 0.8 (the cause of doubling is the presence of two baths). Thus,

$$q_1 V_0(1 + \frac{q_1^*}{q_1} \cdot 0.31) = q_1 V_0(1 + \frac{2 \cdot 0.4}{0.7} \cdot 0.31) = 1.36 \cdot q_1 V_0$$

A taking into account of the spatial distribution enhances the correction factor connected with the lateral dihedral angle only by 5% in comparision to the purely volumetric one.

The coefficient of neutron escape through the lateral sides and lateral edges of the target is equal to $q_4 q_5 = 5.44$.

An estimated of neutron escape through the butt-end side will be discussed below.

Let us consider the inner side of the target as a system of disk sources localized along the beam axis. In the same time these disk sources are substituted by point sources placed in their centres.

A measured distribution of ^{235}U fission fragments track density along the beam axis (fig. 10, R = 10 cm) represents in fact the intensities of these sources. Fig. 14 shows this distribution as a function W(x) where x is the distance from the front border of the target. The graph of the Ω(x) function is shown in fig. 14 as the sum of the solid angles under which the front and back sides (squares 0.5 × 0.5 m^2) of the target. The integral $\int W(x)\Omega(x)dx = 0.13$ represents the quota of neutron escape through the butt-end sides, i.e. $q_6 = \frac{1}{1-0.13} = 1.15(15\%)$.

5 Experimental Results and Discussion

(i) The data of two experiments carried out at the sunchrophasotron 18 GeV ^{12}C+Pb and 3.65 GeV p+Pb are presented in table 1.

One can see from the last column that a good agreement has been received by using the different methods for determinating the neutron

Table 1

	Methods	n_n	V_m/V_d	Σ_H/Σ_d	N_n/Q
	$^{238}U(n,\gamma)$	$1.35 \cdot 10^8$	$0.5 \cdot 10^6$	0.12	$1.1 \cdot 10^{13}$
18 *GeV*	$^{235}U(n,f)$	$0.59 \cdot 10^8$	$1.5 \cdot 10^6$	0.114	$1.0 \cdot 10^{13}$
12 ^{12}C+Pb	$^{23}Na(n,\gamma)$	$2.5 \cdot 10^8$	166	248	$1.0 \cdot 10^{13}$
	$^{238}U(n,\gamma)$	$0.76 \cdot 10^9$	$0.5 \cdot 10^6$	0.172	$0.65 \cdot 10^{14}$
3.65 *GeV*	$^{235}U(n,f)$	$0.30 \cdot 10^9$	$1.5 \cdot 10^6$	0.114	$0.51 \cdot 10^{14}$
p+Pb	$^{23}Na(n,\gamma)$	0.3[illegible] $\cdot 10^9$	336	505	$0.55 \cdot 10^{14}$
	$^{186}W(n,\gamma)$	$0.38 \cdot 10^9$	$5.0 \cdot 10^6$	0.033	$0.63 \cdot 10^{14}$

total number which illustrates high confidence of the measurements.

(ii) From the measured values of q_i we obtain $Q = q_1 \cdot q_2 \cdot \ldots \cdot q_6 = 2.97$. In the experiments with protons the total number of neutrons in the units of Q was determined due to averaging the experimental data obtained by different methods (the last column of table 1). $N_n = (17.2 \pm 2.0) \cdot 10^{13}$, and the number of incident protons was determined by means of Al monitor activity[10]: $N_p = (0.17 \pm 0.02) \cdot 10^{13}$, i.e. neutron yield was found to be equal to $Y = N_n/N_p = (101 \pm 20)$ neutrons per proton.

This result shows that the energy expenditure for neutron generating by protons is equal to (36±8) MeV/neutron. It is higher approximately by 10% than corresponding energy expenditure in the case of light nuclei of the same energy [4]. Nevertheless high productivity of neutron generating by relativistic protons in conjuction with high proton beam intensity achieved at modern synchrotrons and linear accelerators of high energies remains as the most important factor at creation of intensive neutron sources.

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E1-94-407

Измерение нейтронного компонента ливня, индуцированного в свинце релятивистским ядерным пучком

В работе описывается методика определения полного числа нейтронов, образующихся в протяженной свинцовой мишени под действием релятивистских ядер и протонов. Показано, что при энергии протонов 3,65 ГэВ в мишени размером 50×50×80 см образуется 101 ± 20 нейтрон/протон.

Работа выполнена в Лаборатории высоких энергий ОИЯИ.

Препринт Объединенного института ядерных исследований. Дубна, 1994

Chultem D. et al.
E1-94-407

Measurement of the Neutron Component in a Shower Generated in a Lead Target by Relativistic Nuclear Beam

The present paper describes a method of determining the total number of neutrons generated in an extended lead target by relativistic nuclei and protons. It is shown that 101 ± 20 neutrons per proton are produced in the target with the volume of 50×50×80 cm^3 at 3.65 GeV energy of protons.

The investigation has been performed at the Laboratory of High Energies, JINR.

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