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Abstract

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In recent years detailed measurements of neutron transmission functions and of fission self-indication have been carried out for $^{239}$Pu samples of various thicknesses [1], and these measurements have made it possible to test the resonance parameters of $^{239}$Pu. This task is especially urgent in view of the lack of reliable microscopic data for $^{239}$Pu on transmission in the resonance region, which has meant that evaluated resonance parameters [2] could be obtained only on the basis of total cross-section data that are nearly 20 years old [3,4] and for which the experimental resolution function is not accurately known. The direct use in the analysis of experimental data on transmission for a given set of thicknesses can be expected to reduce the uncertainties arising from imprecise knowledge of the resolution function (only with very narrow thicknesses can the resolution function for the total cross-section be obtained from the resolution function for transmission) and from other possible systematic errors, e.g. inaccurately defined sample thicknesses.

As our starting point we selected a system of evaluated resonance parameters [2] which permits the available experimental neutron cross-section data to be described unequivocally in the resonance region. With the aid of this system of parameters we tried to describe as well as possible the detailed behaviour of the transmission and fission self-indication functions for five samples of $^{239}$Pu [1], varying the initial resonance parameters as required.

Van'kov and co-workers [1] measured neutron transmission functions over path lengths of around 1000 and 250 m and fission self-indication over a path length of approximately 78 m. The geometry of the experiment is such that neutron scattering into the detector can be disregarded. Transmission measurements were carried out for five samples varying in thickness from 0.00862 nuclei/b to 0.1294 nuclei/b over the path length of 1000 m (1st series) and for one sample having a thickness of about 0.03 nuclei/b over the path length of 250 m (2nd series) with much better resolution than in the first
series. Fission self-indication was measured for three samples ranging from 0.00217 to 0.0173 nuclei/b. All measurements were made at room temperature; the amount of \(^{240}\text{Pu}\) impurity in the samples was determined from the shape of the experimental curve.

The resonance parameters were tested in three stages. During the first stage the transmission functions for the series of five samples (average path length 1000 \(\pm\) 5 m) were treated simultaneously. The exact path lengths and time lags were calculated for each sample on the basis of the energy scale adopted, referenced to the most clearly defined resonances. For this purpose we took the energy scale of Ribon and Le Cog [5], which is practically the same as the experimental scale [6].

The parameters \(r^0\) and \(\Gamma_a = \Gamma_f + \Gamma_\gamma\) were calculated by means of the Breit-Wigner formalism. For each resonance calculated, the contribution from 50 neighbouring resonances (25 to the right and 25 to the left of the resonance of interest) was taken into account. This made it possible to take into account the contributions to the last resonance treated (at 49.7 eV) from the broad resonances at 96.49 and 100.25 eV; the contribution from the resonance at 100.25 eV amounted to 0.07 b, which indicates that an adequate number of resonances were considered. Moreover, 13 resonances of \(^{240}\text{Pu}\), which were also considered in the treatment, contribute to the region of interest. The resonance parameters of \(^{240}\text{Pu}\) are taken from Ref. [7]. The parameters \(r^0\) and \(\Gamma_a\) for each resonance were varied within certain limits in order to get an optimum description of the curve behaviour for all five samples:

\[
T(E) = \int \exp\left[-x\sigma(E')\right] R(E,E') dE',
\]

where \(x\) is the thickness of the sample; \(\sigma(E')\) is the total cross-section at a given energy point (the temperature dependence is normally taken into account via \(\psi\) and \(\chi\) functions), which depends on the variable parameters; and \(R(E,E')\) is the experimental resolution function. The value of the potential scattering cross-section was taken as 10.35 b [2], which agrees with the data in Ref. [8] (10.30 \(\pm\) 0.15 b) and makes it possible to reconcile the scattering cross-section at the thermal point (7.4 b) with the estimate from Ref. [9] (7.2 \(\pm\) 1.4 b) when parametrizing the cross-sections in the thermal region.
The initial values of the parameters are given in Table 1, together with the quantities \( g_{n}^{0} \) and \( \Gamma_{a} \), obtained after first stage (1st series) fitting. During the second stage, the parameters of the first series were used as base data to describe the shape of the \( T_c(E) \) curve for a sample having a thickness of 0.03019 nuclei/b with a path length of 250 m, measured separately with much better resolution than in the preceding series. The authors did not manage to get into the region above 50 eV because even at this energy level the distance between the experimental points is of the order of a full resonance width (0.2-0.3 eV). The resolution function in this experiment had a more complex form, and the computation time was thereby substantially increased. The system of parameters fitted for the description of the \( T_c(E) \) curve in this sample also appears in Table 1 (2nd series).

Subsequently, calculations of the \( T_c(E) \) curve for the five samples of the first series were carried out using the parameters from series 2 in order to clarify which system of parameters best reproduces the transmissions for these five samples. We shall discuss the need for this procedure later. The calculations revealed that a combination of the parameters of both series (series 3 in Table 1) gives the best results.

Let us now consider some methodological questions arising from the solution of this problem:

1. It turned out to be impossible to carry out a combined evaluation of series 1 and 2 because of the large quantity of experimental information, so the experiments were processed separately.

2. The problem is typical of those requiring the minimization of a function of certain variables:

\[
T_m = \frac{\sum_{k=1}^{n} \sum_{i=1}^{N} T_{eks,k,i} - T_{calc,k,i}}{\Delta T_{eks,k,i}} \rightarrow \text{min}
\]

where \( n \) is the number of samples and \( N \) is the number of experimental points. It turned out that the function in question has local minima in the region of each resonance (in the regions where changes in \( g_{n}^{0} \) and \( \Gamma_{a} \) occur), as a result of which it was not possible to utilize the usual methods of finding the function's minimum (gradient methods etc.). The problem was solved in the following manner: domains of change were established for each of the parameters, and \( T_m \) was computed for each pair of parameters; the pairs were
set up in such a way that each value of one parameter was correlated in turn
with all the values of the other. The "best" pair of parameters, \( r_n^0 \) and \( r_a \),
corresponded to the minimum value of \( T_m \) found in this manner. The certainty
with which the overall minimum is found depends on the size of the parameter
change steps, but if they are kept small enough the increase in computation
time becomes a more significant factor than the accuracy of the parameters
thereby obtained. Although the minimum ascertained may well not be an overall
minimum, it is nevertheless generally "deeper" than the minima yielded by
the gradient and other methods. Now it will be understood that treating
series 1 and 2 separately may result in different values for parameters of one
and the same resonance and, moreover, the resonance parameters of series 2 may
describe the resonances of series 1 better than the actual series 1 parameters
(deeper local minima are achieved in series 2).

Thus, after completing the first and second stages we have a set of
resonance parameters \( \{ r_n^0, r_a \} \), which provide an optimum description of the
experimental data indicated above.

Let us proceed now to the third stage. Since the fission self-indication
function takes the form

\[
T_f(E) = \frac{\int \sigma_f(E') \exp \left[ -x \sigma_f(E') \right] R(E') dE'}{\int \sigma_f(E') R(E') dE'},
\]

where \( \sigma_f(E') \) is the fission cross-section at point \( E' \) and the other quantities
have the same meaning as in expression (1), let us endeavour to solve the
problem of finding the parameters \( \Gamma_f \) and \( \Gamma_\gamma \) for all the resonances under con-
sideration. In fact, the parameters \( r_n^0 \) and \( r_a = \Gamma_f + \Gamma_\gamma \) have already been
obtained, and the spins and resonance energies are also known [2]. If we now
fix \( r_n^0 \) and \( r_a \) and vary \( \Gamma_f \) in such a way that the condition \( \Gamma_a = \Gamma_f + \Gamma_\gamma \) is
fulfilled, i.e. by using only those values of \( \Gamma_f \) which do not take \( \Gamma_\gamma \) beyond
some reasonable interval (see below), then values for \( \Gamma_f \) can be found for
which condition (2) is fulfilled. This would automatically mean that the
values of \( \Gamma_\gamma \) corresponding to them are likewise optimal. Thus this problem
does not, in essence, differ from the one which was solved above. Let us now
consider in detail how to treat the algorithm shown above.
We had at our disposal $T_f(E)$ values measured on three samples with thicknesses ranging from 0.00217 nuclei/b to 0.0173 nuclei/b; the resolution function had the same form as in series 2, and the resolution corresponded approximately to that in series 1. The resonance energies and spins were taken from Ref. [2]; $T_n^0$ and $T_a$ were taken from Table 1 (series 3) and fixed. Furthermore, given the physical meaning of the quantities $T_f$ and $T_\gamma$, the following limitations were imposed on them:

(a) $T_f > 0.1 \times 10^{-2}$ eV; (b) $0.029$ eV < $T_\gamma$ < $0.061$ eV.

By varying $T_f$ within the limits imposed by these conditions and requiring the fulfilment of condition (2), we expected to obtain more precise values of $T_f$ and $T_\gamma$ in the energy region indicated above and thereby to close our system of resonance parameters. It turned out, however, that $T_f(E)$ is not very responsive to changes in the value of $T_f$ within the limits imposed by conditions (a) and (b). Table 2 shows calculations of $T_f(E)$ at experimental energy points very close to the peaks of the corresponding resonances for the various optimal values of $T_f$ obtained by fitting.

The $T_f$ init. in Table 2 are values of $T_f$ obtained from $T_a$ (cf. series 3, Table 1) on the assumption that $T_\gamma = 0.04$ eV (except for those cases where the value of $T_\gamma$ is in brackets); $T_f$ init. are calculated values of fission self-indication for each of the three samples at a given value of $T_f$ init.; $T_f$ are calculated values of $T_f$ obtained by fitting to the experimental data, subject to the condition that the formula for calculating errors in the experimental data has the same form as in series 2. The self-indication values which correspond to these $T_f$ are shown in column $T_f$ init. The notation "max" against $T_f$ denotes that the maximum possible value of $T_f$ was reached if conditions (a) and (b) are taken as the starting point; the notation "mid" denotes that $T_f$ is within the resolved interval. The $T_f$ were obtained by fitting subject to the condition that the error calculation formula has the form $\sim 1/\sigma$, where $\sigma$ is the total cross-section in Breit-Wigner form without allowance for Doppler broadening; this definition of the error means that the experimental points located closer to the resonance peak have the greatest weight, in contrast to the previous formula, where greater weight was attributed to points in the interresonance region.
These two methods of specifying the experimental errors would have been expected to give substantially different fitted values of $\Gamma_f$ in view of the fact that the $T_f(E)$ data used have a very broad scatter in the interresonance region. It can be seen from Table 2 that this is indeed what happened. The notation "min" indicates that the fitting resulted in the smallest possible value of $\Gamma_f$. The values of $T_f$ which correspond to these "min" values are shown in the $T_{f,2}$ column. An analysis of Table 2 reveals that the greatest possible changes in $\Gamma_f$ do not effectively alter the values of $T_f$. This holds true for all resonances except two: those at 41.42 and 44.48 eV, which have large values of $\Gamma^0_n$ compared with the other resonances. The two methods of specifying the experimental errors led to values of $\Gamma_f$ for the corresponding resonances which were at opposite ends of the interval of allowed changes: the maximum values of $\Gamma_f$ were given by the first method owing to the large scatter of $T_f$ in the interresonance region; the minimum values resulted from the second method, except for those cases where there was a very small number of experimental points in the resonance region (for example at 49.71 eV) and where the resonance does not in effect emerge from the shape of the curve (for example at 11.5 eV).

It is thus evident that the experimental data on fission self-indication do not permit us to ascertain or test the values of the fission widths owing to the very low sensitivity of $T_f$ to changes in the broad limits of $\Gamma_f$. However, using transmissions measured for various thicknesses we can obtain the resonance parameters $\Gamma^0_n$ and $\Gamma$ by the form method, where the effect of insufficiently precise measurement of the resonance peaks is not so telling. We may conclude that the experimental data of Vankov and co-workers [1] do not contradict the available microscopic data for $\sigma_t$ and enable us to obtain more accurate values of $g\Gamma_n$ for $^{239}$Pu in the energy region below 50 eV. The currently available resonance parameters could be rendered very much more precise if experiments were carried out utilizing large sets of sample thicknesses with cooling and with good resolution for the various fissile nuclei.
REFERENCES

[1] VAN'KOV, A.A., GRIGOR'EV, Yu.V., UKRAINTSEV, B.F. et al., An experimental study of resonance self-shielding of the total cross-section and fission cross-section for $^{239}$Pu, Voprosy atomnoj nauki i tekhniki (Questions of atomic science and technology) Nuclear Constants Series No. 2(37) (1980) 44-50.

[2] ANTSIPOV, G.V., BACHANOVICH, L.A., ZHARKOV, V.F. et al., Evaluation of nuclear data for $^{239}$Pu, Part I: The resolved resonance region, $10^{-5}$ - 660 eV, ITMO Preprint, Byelorussian Academy of Sciences, Minsk (1981) 1-48.

[3] DERRIEN, H., BLONS, G., EGGERMAN, C. et al., Sections efficaces total et de fission du $^{239}$Pu. In: Proc. IAEA Conf. on Nuclear Data for Reactors (Paris, 1966). Vienna, IAEA, 1967, v. 2, p. 195-219.

[4] BOLLINGER, L.M., COTE, R.E., THOMAS, G.E., Low-energy total and fission cross-section measurements of $^{239}$Pu. In: Proc. of the Intern. Conf. on Peaceful Uses of Atomic Energy. Geneva, 1958, v. 15, p. 127-135.

[5] RIBON, P., LE COG, G., Evaluation des Donnees Neutroniques de $^{239}$Pu. CEA-N-1484. France, Saclay, 1971.

[6] GWIN, R., WESTON, L.W., DE SAUSSUVE, G. et al., Simultaneous measurement of the neutron fission and absorption cross-sections of $^{239}$Pu over the energy region 0.02 eV to 30 keV. Nucl. Sci. and Engng, 1971, v. 45, p. 25-36.

[7] ANTSIPOV, G.V., KON'SHIN, V.A., SUCHOVITSKIJ, E.Sh., Nuclear constants for plutonium isotopes, Minsk (1981).

[8] UTTLEY, C.A., Neutron total cross-section measurements. Progress Report AERE-PR/N, 1967, p. 11; EANDC(UK)-35B, 1964.

[9] LEMMEL, H.D., The third IAEA evaluation of the 2200 m/s and 20°C Maxwellian neutron data for $^{233}$U, $^{235}$U, $^{239}$Pu, $^{241}$Pu. In: Proc. of the Conf. on Nucl. Cross-Sections and Technology. Washington, 1975, NB5, Spec. Public. 425, v. 1, p. 286-292.

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### Comparison of resonance parameters in the different series

| $E_z$ | $g'_{n\text{ init.}}$ | $g'_{n \text{ s.1}}$ | $g'_{n \text{ s.2}}$ | $g'_{n \text{ s.3}}$ | $g'_{n \text{ s.3}}$ | $g'_{n \text{ s.3}}$ | $\Delta g'_{n\text{ init.}}$ | $\Delta g'_{n}$ | Notes |
|-------|------------------------|----------------------|----------------------|----------------------|----------------------|----------------------|------------------|-----------------|-------|
| 5.9   | 4.7·10^{-3}            | 3.3023               | 3.3023               | 3.3023               | 3.3023               | 3.3023               | 0                | 0               | Init. |
| 7.82  | 5.7351·10^{-4}         | 9.7·10^{-2}          | 8.497·10^{-2}        | 6.195·10^{-4}        | 7.878·10^{-4}        | 5.413·10^{-2}        | 8.497·10^{-2}    | 5.12            | 3.34  |
| 10.93 | 1.3239·10^{-3}         | 1.998·10^{-1}        | 1.668·10^{-1}        | 1.266·10^{-3}        | 1.491·10^{-1}        | 1.196·10^{-3}        | 1.668·10^{-1}    | 9.59            | 16.06 |
| 12.5  | 4.2527·10^{-5}         | 5.16·10^{-2}         | 1.569·10^{-1}        | 4.85·10^{-5}         | 1.569·10^{-1}        | 4.872·10^{-5}        | 1.569·10^{-1}    | -14.58          | -20.413|
| 11.89 | 6.6947·10^{-4}         | 7.6·10^{-2}          | 6.45·10^{-2}         | 5.791·10^{-4}        | 5.805·10^{-2}        | 5.791·10^{-4}        | 5.805·10^{-2}    | 13.49           | 23.60 |
| 14.31 | 4.3221·10^{-4}         | 7.81·10^{-2}         | 1.012·10^{-1}        | 9.95·10^{-2}         | 7.181·10^{-4}        | 9.95·10^{-2}         | 7.181·10^{-4}    | -66.15          | -0.28  |
| 14.68 | 1.4198·10^{-3}         | 6.8·10^{-2}          | 2.078·10^{-3}        | 3.43·10^{-2}         | 6.173·10^{-2}        | 3.43·10^{-2}         | 6.173·10^{-2}    | -0.26           | 7.13   |
| 15.46 | 4.6707·10^{-4}         | 9.89·10^{-1}         | 6.989·10^{-1}        | 5.328·10^{-4}        | 6.596·10^{-1}        | 6.329·10^{-4}        | 6.596·10^{-1}    | 0.06            | 0     |
| 17.66 | 1.2249·10^{-3}         | 7.3·10^{-2}          | 7.3·10^{-2}          | 1.49·10^{-3}         | 6.328·10^{-1}        | 1.49·10^{-3}         | 6.328·10^{-1}    | 0.17            | 0     |
| 22.29 | 1.8573·10^{-3}         | 1.06·10^{-1}         | 2.027·10^{-3}        | 2.231·10^{-3}        | 8.644·10^{-2}        | 2.231·10^{-3}        | 8.644·10^{-2}    | -9.15           | 9.73   |
| 23.94 | 6.386·10^{-5}          | 7.1·10^{-2}          | 6.945·10^{-4}        | 6.250·10^{-5}        | 5.739·10^{-2}        | 5.250·10^{-5}        | 5.739·10^{-2}    | 2.12            | 18.02  |
| 26.24 | 8.9929·10^{-4}         | 8.2002·10^{-2}       | 8.466·10^{-2}        | 1.240·10^{-3}        | 7.718·10^{-2}        | 1.240·10^{-3}        | 7.718·10^{-2}    | -25.4           | -4.22  |
| 27.24 | 1.0735·10^{-4}         | 4.2·10^{-2}          | 1.029·10^{-3}        | 9.183·10^{-2}        | 3.249·10^{-5}        | 9.183·10^{-2}        | 3.249·10^{-5}    | 14.41           | 22.63  |
| 32.31 | 1.8974·10^{-4}         | 1.51·10^{-1}         | 1.877·10^{-4}        | 1.764·10^{-1}        | 1.764·10^{-1}        | 1.764·10^{-1}        | 1.764·10^{-1}    | 5.88            | 9.46   |
| 34.6  | 9.16·10^{-6}           | 9.1·10^{-2}          | 8.24·10^{-6}         | 8.219·10^{-2}        | 8.24·10^{-6}         | 8.219·10^{-2}        | 8.24·10^{-6}     | 10.0            | 9.68   |
| 35.5  | 2.0414·10^{-4}         | 4.7·10^{-2}          | 2.0458·10^{-4}       | 1.841·10^{-4}        | 1.841·10^{-4}        | 1.841·10^{-4}        | 1.841·10^{-4}    | 9.92            | 9.83   |
| 41.42 | 3.1786·10^{-3}         | 4.8·10^{-2}          | 3.2805·10^{-3}       | 2.934·10^{-3}        | 4.806·10^{-3}        | 2.934·10^{-3}        | 4.806·10^{-3}    | 7.69            | -0.01  |
| 41.66 | 1.1150·10^{-3}         | 1.04·10^{-1}         | 9.3891·10^{-4}       | 9.796·10^{-2}        | 9.796·10^{-2}        | 9.796·10^{-2}        | 9.796·10^{-2}    | 14.11           | 24.14  |
| 44.48 | 4.7089·10^{-3}         | 5.2·10^{-2}          | 4.0079·10^{-3}       | 3.9964·10^{-2}       | 3.7626·10^{-2}       | 3.9964·10^{-2}       | 3.7626·10^{-2}   | 14.88           | 23.15  |
| 47.6  | 1.4101·10^{-3}         | 3.06·10^{-1}         | 1.2087·10^{-3}       | 1.3299·10^{-3}       | 2.1149·10^{-1}       | 1.2087·10^{-3}       | 2.1149·10^{-1}   | 14.28           | 23.91  |
| 49.71 | 1.0143·10^{-3}         | 7.98·10^{-1}         | 9.2044·10^{-4}       | 9.2145·10^{-1}       | 9.2044·10^{-4}       | 9.2145·10^{-1}       | 9.2044·10^{-4}   | 9.29            | -12.38 |
Values of the quantities $r_f$ and $T_f$

| $E_z$ | $r_f$ init. | $T_f$ init. | $r_{f1}$ | $T_{f1}$ | $r_{f2}$ | $T_{f2}$ |
|------|-------------|-------------|----------|----------|----------|----------|
| 5.9  | 0.3259·10^{-1} | 0.958429 | 0.32573·10^{-1} | 0.95843 | 0.32493·10^{-1} | 0.95852 |
|      | ($r_f=0.0433$) | 0.841212 | 0.84121 | 0.70710 | 0.84153 | 0.70764 |
| 7.82 | 0.44097·10^{-1} | 0.54994·10^{-1} | 0.55097·10^{-1} | 0.54994·10^{-1} | 0.23097·10^{-1} | 0.54964·10^{-1} |
|      | ($r_f=0.045$) | 0.46175·10^{-3} | 0.46152·10^{-3} | 0.23097·10^{-3} | 0.46186·10^{-3} | 0.16792·10^{-2} |
| 10.93 | 0.12683 | 0.10133 | 0.10133 | 0.10583 | 0.10583 | 0.10583 |
|      | 0.706139·10^{-1} | 0.70814·10^{-4} | 0.70814·10^{-4} | 0.70814·10^{-4} | 0.17591·10^{-1} | 0.17591·10^{-1} |
| 11.50 | 0.12638 | 0.12971 | 0.12793 max | 0.12793 max | 0.12793 max | 0.12793 max |
| 11.89 | 0.18065·10^{-1} | 0.244769 | 0.25065·10^{-1} max | 0.24379 | 0.17793·10^{-2} | 0.17793·10^{-2} |
|      | 0.94104·10^{-2} | 0.22372·10^{-4} | 0.22372·10^{-4} | 0.22372·10^{-4} | 0.17694·10^{-1} | 0.17694·10^{-1} |
| 14.31 | 0.6128·10^{-1} | 0.171693 | 0.7228·10^{-1} max | 0.17173 | 0.4029·10^{-1} min | 0.4029·10^{-1} min |
|      | 0.25162·10^{-2} | 0.25162·10^{-2} | 0.25162·10^{-2} | 0.25162·10^{-2} | 0.16687 | 0.16687 |
| 14.68 | 0.23153·10^{-1} | 0.23153·10^{-1} | 0.34153·10^{-1} max | 0.23074·10^{-1} | 0.51524·10^{-5} | 0.51524·10^{-5} |
|      | 0.52617·10^{-5} | 0.29505·10^{-8} | 0.29505·10^{-8} | 0.29505·10^{-8} | 0.7331·10^{-5} | 0.7331·10^{-5} |
| 15.46 | 0.6489 | 0.74776 | 0.6699 max | 0.6699 max | 0.6699 max | 0.6699 max |
|      | ($r_f=0.05$) | 0.3147 | 0.3147 | 0.97927·10^{-1} | 0.97927·10^{-1} | 0.97927·10^{-1} |
| 17.66 | 0.3240·10^{-1} | 0.12773 | 0.440·10^{-1} max | 0.12772 | 0.120·10^{-1} min | 0.120·10^{-1} min |
|      | 0.26904·10^{-3} | 0.26775·10^{-3} | 0.26775·10^{-3} | 0.26775·10^{-3} | 0.26989·10^{-3} | 0.26989·10^{-3} |
| 22.29 | 0.55687·10^{-1} | 0.84910·10^{-1} | 0.66687·10^{-1} max | 0.84905·10^{-1} | 0.34687·10^{-2} min | 0.34687·10^{-2} min |
|      | 0.61242·10^{-3} | 0.61223·10^{-3} | 0.61223·10^{-3} | 0.61223·10^{-3} | 0.61334·10^{-3} | 0.61334·10^{-3} |
|      | 0.11592·10^{-3} | 0.11990·10^{-3} | 0.11990·10^{-3} | 0.11990·10^{-3} | 0.11630·10^{-3} | 0.11630·10^{-3} |
| $E_z$  | $\Gamma_1$ | $T_1$ | $\Gamma_2$ | $T_2$ |
|-------|----------|-------|----------|-------|
| 23.94 | 0.17387·10^{-1} | 0.900965 | 0.89990 | 0.91490 |
|       | 0.663048 | 0.65972 | 0.67050 | 0.70855 |
| 26.24 | 0.45466·10^{-1} | 0.354078 | 0.35406 | 0.35417 |
|       | 0.24146·10^{-1} | 0.24128·10^{-1} | 0.24200·10^{-1} |
| 27.24 | 0.2494·10^{-2} | 0.89344 | 0.89107 | 0.89102 |
|       | (\Gamma = 0.03) | 0.663076 | 0.67050 | 0.66933 |
| 32.31 | 0.9671·10^{-1} | 0.876109 | 0.87603 | 0.35431 |
|       | 0.612515 | 0.61234 | 0.61305 |
| 34.6  | 0.42195·10^{-1} | 0.973733 | 0.97370 | 0.97368 |
|       | 0.828261 | 0.82828 | 0.82879 |
| 35.5  | 0.2379·10^{-2} | 0.906876 | 0.90262 | 0.90253 |
|       | 0.594741 | 0.64299 | 0.64268 |
| 41.42 | -0.8006·10^{-2} | 0.312426 | 0.31267 | 0.31529 |
|       | 0.16179·10^{-2} | 0.16364·10^{-2} | 0.16389·10^{-2} |
| 41.66 | 0.38895·10^{-1} | 0.550427 | 0.55429 | 0.55418 |
|       | 0.41097·10^{-1} | 0.41261·10^{-1} | 0.41305·10^{-1} |
| 44.48 | 0.9964·10^{-2} | 0.328434 | 0.32769 | 0.37812 |
|       | (\Gamma = 0.03) | 0.31038·10^{-1} | 0.30772·10^{-1} | 0.30772·10^{-1} |
| 47.6  | 0.19284 | 0.805163 | 0.80484 | 0.80576 |
|       | 0.367239 | 0.36673 | 0.36673 |
| 49.71 | 0.65692 | 0.29853 | 0.29665 | 0.29395 |
|       | 0.38134·10^{-1} | 0.39279·10^{-1} | 0.39279·10^{-1} |

Table 2 (continued)
It is generally known that $^{239}\text{Pu}$ and $^{233}\text{U}$ are considered to be the basic fuel for nuclear power plants utilizing fast reactors. A comprehensive study of the nuclear physics characteristics of these isotopes is accordingly of great interest. The fission neutron spectrum is one of these characteristics and a substantial number of papers has been devoted to it. Most of them, however, deal with the energy region above 1 MeV. The neutron energy range between 0 and 1 MeV, which accounts for some 25% of fission spectrum neutrons, has had only one paper [1] devoted to it. The conclusions of this study, together with the relevant data on $^{235}\text{U}$ [1, 2] and $^{252}\text{Cf}$ [1, 3-7], indicate that the method currently used in reactor calculations to obtain information on the neutron spectrum for this energy range (extrapolation from the higher energy range using the Maxwell or Watt formulae) can at present hardly be considered adequately substantiated.

Our present work was aimed at studying prompt neutron spectra in the 10 keV to 3 MeV range for thermal neutron fission of $^{239}\text{Pu}$ and $^{233}\text{U}$, using the time-of-flight and the lithium glass method. The measurements were carried out in the tangential channel of the VVRS-M reactor belonging to the Central Institute of Physical Research. The beam was shaped by a special collimator, fast neutrons and γ-quanta being filtered out by means of quartz and polycrystalline bismuth plugs. Figure 1 shows a diagram of the experiment. A scintillation chamber filled with pure argon at atmospheric pressure served as a fission fragment detector. The chamber was so designed that a target prepared from the material being studied and one made of $^{252}\text{Cf}$ having approximately the same characteristics could be introduced alternately into the working volume without adversely affecting the functioning of the apparatus or the geometry of the experiment. In order to optimize the ratio of pulse amplitudes between fission fragments and α-particles (this is particularly important in the case of measurements involving $^{239}\text{Pu}$), a quartz plate 65 mm in diameter and 0.6 mm thick was positioned at a distance of 9 and 41 mm from the layer of fissile material.
and the photomultiplier photocathode respectively. The diameters of all the
layers comprised 45 mm. Stainless steel foil 0.1 mm thick served as a
support. The layers of $^{239}$Pu and $^{233}$U were of equal thickness, 1.0 mg/cm$^2$.
The intensity of the fission fragments in the measurements of $^{239}$Pu, $^{233}$U
and $^{252}$Cf was $6.4 \times 10^3$, $5.1 \times 10^3$ and $2.7 \times 10^4$ s$^{-1}$ respectively.

Number 912 and No. 913 lithium glasses 45 mm in diameter and 9.5 mm thick
served as neutron and $\gamma$-ray detectors respectively. In order to reduce
the background of random coincidences, the neutron and $\gamma$-ray detectors
were placed within a special shield composed of paraffin, $^6$Li hydride and
lead. The path length, the analyser channel width and the total prompt
$\gamma$-ray peak width at half height were 30.5 cm, 0.478 ns and 3 ns respectively.
The experimental method is described in Ref. [2].

The measurement procedure for each isotope studied consisted of eight
sequentially executed series: $^{239}$Pu ($^{233}$U) - NE-912, $^{239}$Pu ($^{233}$U) - NE-912
(with cone), $^{239}$Pu ($^{233}$U) - NE-913, $^{239}$Pu ($^{233}$U) - NE-913 (with cone) and
$^{252}$Cf - NE-912, $^{252}$Cf - NE-912 (with cone), $^{252}$Cf - NE-913, $^{252}$Cf - NE-913
(with cone). Aggregate quantities of detected fragments for these series are
given in the table. Figure 2 depicts the measured spectra on a time scale.
In order to ascertain the correction required for the finite geometry of
the experiment, we performed, when measuring the scattered neutron background,
an additional experiment using layers of californium 7 and 45 mm in diameter.

In processing the data we applied a relative method which relies on
the prompt neutron spectrum from spontaneous fission of $^{252}$Cf as a standard.
In the first stage all the distributions measured were normalized to a
single neutron emitted during fission. Then the background of random and
systematic random coincidences was subtracted. Next, the time spectra
measured with the NE-913 glass were subtracted from the spectra measured
with the NE-912 glass, i.e. the delayed $\gamma$-background was taken into account.
After this the scattered neutron background was subtracted. Note that,
strictly speaking, measurements of the scattered neutron background by
the shadow cone method are justified only where a point source is used.
Here this was not the case, and accordingly the background measured during
the experiment is somewhat understated. The ratio between two time spectra
measured in the one case with a small (7 mm diameter) and in the other
case a large (45 mm diameter) californium target and the corresponding
shadow cones (copper, 12 cm in length) was used to correct for this effect.
It is evident that, given ideal "absolutely black" cones, this ratio will be determined solely by the geometry of the experiment and should not depend on the neutron energy. In our experiment this correction turned out to be 1.20, and it did not change by more than ± 5% in the energy range which was being studied.

Subsequently the distributions obtained were converted to an energy scale, the neutron detection efficiencies were determined (from measured $^{252}$Cf spectra, on the assumption that they can be described by a single-parameter distribution with $T = 1.42$ MeV), and prompt neutron energy spectra for thermal fission of $^{239}$Pu and $^{233}$U (Figs 3, 4) were ascertained.

The results of extrapolating data from the higher energy region by means of single-parameter Maxwell distributions, with $T = 1.38$ MeV for $^{239}$Pu and 1.32 MeV for $^{233}$U, are also shown in these two figures.

It will be seen that, within the experimental error limits (approx. 5%) the neutron spectra of both isotopes in the 10 keV to 3 MeV range are satisfactorily described by these distributions. For $^{239}$Pu, this result is consistent with the corresponding data in Ref. [1]. For $^{233}$U, however, there is some divergence. In contrast to the present authors' findings, the results of Ref. [1] indicate a $^{233}$U fission neutron spectrum 5 to 10% above the Maxwellian distribution in the 0.05-0.3 MeV range.

In conclusion it should be remarked once again that the data in the experiment we have described was treated by a relative method. For this reason the results it yielded for $^{239}$Pu and $^{233}$U may be interpreted as follows: if the prompt neutron spectrum from $^{252}$Cf spontaneous fission in the 10 keV to 3 MeV range is described by a Maxwellian distribution with $T = 1.42$ MeV, then the neutron spectra for $^{239}$Pu and $^{233}$U in this energy region exhibit no deviations (beyond ± 5%) from Maxwellian distributions with $T = 1.38$ and $T = 1.32$ MeV respectively (cf. the solid curves in Figs 3 and 4 respectively).
REFERENCES

[1] NEFEDOV, V.N., STAROSTOV, B.I., SEMENOV, A.F., In: Nejtronnaya Fizika (Neutron Physics) (Proceedings 4th All-Union Conference on Neutron Physics, Kiev, 18-22 April 1977), Moscow, TsNIIatominform Part 3, (1977), 205; preprint of NIIAR P-22 (356) (1978).

[2] LAJTAI, A., KECHKEMETI, J., KLUGE, D. et al., Ibid, 26.

[3] MEADOWS, J.W., Phys. Rev., 1967, v. 157, p. 1076.

[4] ZAMYATHIN, J.S., KROSHKIN, N.I., MELNIEKOV, A.N., NEFIEDOV, W.N. et al., Nuclear Data for Reactors, Vienna: IAEA, 1970, v. 11, p. 183.

[5] JEKI, L., KLUGE, C., LAJTAI, A. et al., Prompt Fission Neutron Spectra, Vienna: IAEA, 1972, p. 81.

[6] D'YACHENKO, P.P., SEREGINA, E.A., KUTSAEVA, L.S. et al., Atomnaya Ehnergiya 42 (1977) p. 25.

[7] BLINOV, M.V., VITENKO, V.A., TOUSE, V.T., Neutron Standards and Application, NBS Special publ. 493, Washington, 1977, p. 194.

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**Fig. 1.** Diagram of the experiment

1. Fission fragment detector
2. Photomultiplier 56-UVP
3. Quartz
4. 235U target
5. 252Cf target
6. Paraffin
7. Lithium glass
8. Photomultiplier 56-UVP
9. Lead
10. Li hydride
11. Neutron (γ-ray detector)
12. Discriminator with follow threshold
13. Delay line
14. Time-amplitude converter
15. Analyser
16. Coincidence circuit
17. Differential discriminator
18. Amplifier
Numbers of fragments recorded (x 10^9)

| Neutron (γ-ray) detectors | 239Pu | 233U | 252Cf |
|--------------------------|-------|------|-------|
| NE-912                   | 7,789 | 10,800 | II,065 |
| NE-913                   | 7,734 | 12,713 | 12,254 |
| (with cone)              | 9,114 | 8,600  | 19,212 |
| NE-912                   | 5,95  | 12,229 | 37,090 |
| (with cone)              |       |       |       |

Fig. 2.: Measured spectra on a time scale:
1 - NE-912; 2 - NE-913; 3 - NE-912 (copper);
4 - NE-913 (copper)
Fig. 3.: Prompt neutron spectrum from thermal fission of $^{239}$Pu.

Fig. 4.: Prompt neutron spectrum from thermal fission of $^{233}$U.
OPTIMIZATION OF DELAYED NEUTRON DETECTION UNITS FOR ACTIVATION ANALYSIS OF URANIUM AND THORIUM

E.G. Bertman

Activation analysis of geological materials for uranium and thorium by the delayed neutron method makes special demands on delayed neutron detection units. Increasing the sensitivity of the method requires maximum neutron detection efficiency and minimum background. The six basic groups of delayed neutrons have a half-life of 0.2-1 min. They are therefore detected shortly after irradiation of the sample in the reactor, usually in 10-20 s. In this situation the residual induced γ-activity in the matrix of the geological sample (1-20 g per sample) can be very significant and attain 1000 R/h, a feature which must be allowed for in selecting the neutron detector.

Fission chambers detect neutrons with high selectivity against a background of γ-ray fluxes of virtually any intensity, but have low sensitivity to neutrons. Fast delayed neutrons with a mean energy of 0.49 MeV can easily be moderated by placing the irradiated sample in a moderating medium such as paraffin or water. Slow neutrons, on the other hand, can be detected with extreme sensitivity and efficiency by gas discharge counters with a boron-coated cathode or filled with $^3$He [1]. Among the gas discharge counters having a high permissible γ-background value (up to 1000 R/h), the SNM-11 corona counter with its 15.5% neutron detection efficiency has the best parameters.

When analysed for thorium, the sample is irradiated by a flux of fast neutrons only, which yields induced γ-activity two orders of magnitude lower than in the analysis for uranium when the sample is irradiated with thermal neutrons. In this case helium counters with a maximum permissible γ-background of 100 and 20 R/h should be used, i.e. types SNM-17 and SNM-18 respectively, the detection efficiency of which attains 80%. The delayed neutron detection unit appears as a cylinder of the moderating medium in the centre of which the sample is positioned after being transported out of the reactor by a pneumatic conveyor. Gas discharge counters are positioned round the channel and sample in a circle of radius R. The optimum number of counters and their distance from the sample can be determined simply and
accurately by experiment in such a way that the counting efficiency of the unit is maximized. The radial distribution of the neutrons from the (Po-α-Be) source in water, paraffin and graphite was measured for different numbers of counters from 1 to 50. One of these counters was used to detect the neutrons, and from this it was possible to determine the optimum radius (7 cm) of the circle on which the counters were placed around the source. The dimensions of the detection unit are determined by the length of the counter and the distribution of neutrons in the moderator. For detection units using water, paraffin or polyethylene as moderator and utilizing counters of types SNM-11, SNM-17 and SNM-18, the optimum height and diameter are 40–50 cm.

The neutron detection efficiency increases with the number of counters used, the number being restricted in the case of short radii by the size of the counters themselves. Positioning the counters close to one another in rows results in displacement of the moderator, and creates a neutron flux depression which reduces the counting efficiency. The authors of Ref. [2] recommend an optimum detection unit with ten counters positioned in a circle with a diameter of 20 cm. From our own experiments we conclude that for moderators containing hydrogen the optimum number of counters is 20. Further increasing their number does not achieve any material increase in the neutron count rate N and merely complicates and raises the cost of the detection unit and its electronic circuitry. In Table 1 we compare detection efficiencies \( \varepsilon \) for units with different moderators and 20 counters of types SNM-11 and SNM-18. It is evident that a unit using water as moderator and counters of type SNM-18 has the highest efficiency - 33%.

Another means of increasing the sensitivity of the delayed neutron method is to use a subcritical breeder assembly to amplify the low neutron fluxes emitted by the sample. The experiments were carried out for a breeder with a heterogeneous core whose dimensions, like the mass of the fissile material, were well below critical. The multiplication constant was considerably less than unity, which simplified the design and control of the assembly. The measurements were carried out in a geometry with 20 type SNM-11 counters installed in the water moderator around a circle with a radius of 7 cm. A \( ^{252} \text{Cf} \) neutron source was placed in the centre, the source being surrounded (R = 4.4 cm) by a cylindrical multiplier of variable thickness consisting of uranium enriched to 95% in \( ^{235} \text{U} \).
The multiplication constants of the detected neutrons as a function of 
$^{235}$U thickness are shown below:

| Thickness, mm: | 0   | 0.4 | 0.8 | 1.2 | 1.6 |
|---------------|-----|-----|-----|-----|-----|
| $C_m$         | 1.0 | 1.48| 1.50| 1.58| 1.68|

It follows from these data that the multiplication constant for the neutrons 
detected, $C_m$, which is equal to the ratio between the count rate with a 
multiplying medium and the count rate without, rises to 1.48 even for a 
$^{235}$U thickness of 0.4 mm. Increasing the thickness of the multiplier further 
has much less effect: at a thickness of 1.6 mm we find $C_m = 1.68$.

The detection efficiency can be raised by 68% if we use a delayed neutron 
detection unit with a heterogeneous multiplier. A multiplier with a homo-
geneous zone enables the efficiency to be increased by 2-3 orders of magnitude, 
as rough calculations have shown. It must, however, be borne in mind that 
using a multiplier in the detection unit increases the numbers of useful and 
of background neutrons in equal measure. Since the neutron multiplication 
effect results from fissions in the multiplier substance, the emission of 
delayed neutrons is also a consequence of this process. The latter have a 
half-life comparable to the detection time in analysing samples for uranium 
and thorium. This can lead to an accumulation factor which influences the 
count rate when only brief intervals occur between two or three consecutive 
analyses.

Table 2 gives measurements of the delayed neutron background from a 
multiplier consisting of 95% enriched $^{235}$U (a cylinder with $h = 50$ cm, 
$R = 4.4$ cm, $d = 2.4$ mm) after irradiation with neutrons from a $^{252}$Cf source 
($\Phi = 1.45 \times 10^7$ n s$^{-1}$) for 60 seconds without additional shielding against 
external background neutrons. These results support the assumptions expressed 
above. However, the ratio of background delayed neutrons to the number of 
neutrons detected from the source is rather low ($8 \times 10^{-4}$). Since in analyses 
of natural objects the intensity of the delayed neutrons emitted by the sample 
is much lower than in the case of the $^{252}$Cf source used in the experiments, 
the contribution of background delayed neutrons will be so small that it may 
be ignored altogether.
Experiments with delayed neutrons have indicated a detection efficiency 5.5 times greater than for neutrons from $^{252}$Cf or Pu-$\alpha$-Be sources. The results for units with a paraffin moderator and 20 counters of types SNM-11, SNM-17 and SNM-18 are given in Table 3.

A full-scale set-up for analysing geological samples by the delayed neutron method should be equipped with two detection units: type SNM-11 without a multiplier for uranium analysis and SNM-18 with a multiplier for thorium analysis.

REFERENCES

[1] KOROLEV, V.I., CHAJKOVSKIJ, V.G., Pribory i tekhnika ehksperimenta (Experimental instruments and techniques) No. 4 (1969) 208.
[2] VOROB'EV, Yu.A., UGOL'TSEV, O.A., In: Transactions of the Second All-Union Conference on the Metrology of Neutron Radiation in Reactors and Accelerators, Vol. 1, Moscow (1974) 90.

The article was submitted for publication on 7 February 1983.
Table 1. Detection efficiency for a neutron flux with $4.8 \times 10^3 \text{n s}^{-1}$ from a (Pu-α-Be) source, obtained with a unit using 20 SNM-11 and SNM-18 counters in various moderators

| Moderator dimensions                  | Optimiz. radius, cm | SNM-11 | SNM-18 |
|--------------------------------------|---------------------|--------|--------|
|                                      |                     | $N, \text{c/s}$ | $\varepsilon, \%$ | $N, \text{c/s}$ | $\varepsilon, \%$ |
| Water, 50 x 50 cm                    | 7                   | 230    | 4.8    | 1580   | 33     |
| Paraffin, 50 x 50 x 50 cm            | 7                   | 220    | 4.6    | 1440   | 30     |
| Paraffin with bismuth shield, 60 x 60 x 60 cm | 11                 | 153    | 3.2    | 1150   | 24     |
| Polyethylene, 40 x 40 x 40 cm        | 7.5                 | 216    | 4.5    | 1390   | 29     |
| Graphite, 40 x 40 x 40 cm            | 12                  | 48     | 0.1    | 62     | 1.3    |

Table 2. Dependence of multiplier delayer neutron background on measurement time

| Background, c/s | Seconds after irradiation |
|-----------------|---------------------------|
|                 | 10 | 20 | 30 | 40 | 50 | 60 |
| With multiplier | 78 | 30 | 16 | 10 | 8  | 6  |
| Without multiplier | 7   | 8  | 6  | 8  | 7  | 6  |

Table 3. Parameters of neutron detection units

| Parameters                              | SNM-11 | SNM-17 | SNM-18 |
|-----------------------------------------|--------|--------|--------|
| Delayed neutron detection efficiency, %|        |        |        |
| Without multiplier                      | 25     | 64     | 78     |
| With multiplier                         | 42     | 108    | 131    |
| Permissible background of $\gamma$-radiation | 1000  | 100    | 20     |
Reference [1] presents 188 experimental yields for 140 radioactive nuclides, obtained by irradiating thick targets of various chemical elements with 22 MeV protons. The reactions which produced the nuclides are also given: \(\text{pn, p2n, p3n, p(pn+2n), p2p, p3n, pα, pαn, and so on.}\) The large volume of data on nuclide yields presented in Ref. [1] allows us to compare yields from reactions of various types with yields from reactions of a single type for a broad range of nuclear masses.

The yield of a reaction can be defined as the quantity of reaction events per 1000 bombarding particles. For each event of a given reaction, an atom of a certain radionuclide is formed. If irradiating a chemical element produces a radionuclide through only one reaction, the reaction yield can be ascertained from the radionuclide yield by the following formula:

\[
W = 0.554Z^a T_{1/2} (B/P_c),
\]

where \(W\) is the reaction yield in atoms per 1000 particles; \(Z^a\) is the relative charge of the bombarding particle; \(T_{1/2}\) is the nuclide's half-life in days; \(B\) is the radionuclide yield in MBq/(μA*h); \(P_c\) is the content of target isotope, in %. The following formula links the reaction yield with the effective reaction cross-section averaged over the mean free path of the incident particle:

\[
\overline{\sigma} = 1.66 \times 10^3 W(A/R),
\]

where \(\overline{\sigma}\) is the effective cross-section, in mb; \(A\) is the relative atomic mass of the element; and \(R\) is the mean free path of the particle, in mg/cm\(^2\).

From formula (1), using the quantity \(P_c\), we can determine both the reaction yield and the reaction cross-section for a target isotope concentration of 100%.

In this paper most of the radionuclide yields presented in Ref. [1] have been converted to yield values for nuclear reactions of a particular kind and have been classified according to the type of reaction. After Ref. [1] had been submitted for printing the yields of eight radionuclides were measured: \(^{45}\text{Ti}(\text{Sc}), \(^{87m}\text{Y}(\text{Sr}), \(^{87}\text{Y}(\text{Zr}), \(^{89}\text{Zr}(\text{Zr}); \(^{91m}\text{Nb}(\text{Zr}), \(^{117}\text{Sn}(\text{Sn}), \)
$^{117}\text{Sb(Sn)}, \; ^{193}\text{Au(Pt)}$ for $E_p = 22$ MeV (the irradiated elements are shown in brackets). These yields have also been converted to reaction yields. The results of the calculations are given in the table. Yields were determined for 185 reactions in all (the number of reactions is given in brackets) of the following types: $\text{pn}$ (59), $\text{p}2\text{n}$ (44), $\text{ppn}$ (23), $\text{p(}\text{pn+2n})$ (13), $\text{p}3\text{n}$ (13), $\text{p}\alpha$ (7), $\text{p}2\text{p}$ (6), $\text{p}\alpha$ (6), $\text{p(}\text{pn+2p})$ (5), $\text{p}\alpha$ (6), $\text{pt}$ (5), $\text{p(}\text{t+3n})$ (2), $\text{p}\alpha$ (1), $\text{p}2\text{n}$ (1).

As can be seen from Ref. [1] (Table 1, column 3), radionuclides result from one or several reactions. If a radionuclide results from a single reaction, the reaction yield is determined by formula (1). For example, if copper is irradiated the $^{65}\text{Zn}$ yield is 0.59 MBq/(μA·h), the reaction of interest is $^{65}\text{Cu(pn)}$ $^{65}\text{Zn}$, $P_c = 30.9\%$, and the reaction yield given by Eq. (1) is $W = 2.6$ atoms/1000 protons. Reactions resulting in a cumulative yield have the same nucleus target, and the reaction yield is calculated from Eq. (1). For example, if germanium is irradiated the cumulative yield of $^{69}\text{Ge}$ from the $^{70}\text{Ge(ppn)}$ $^{69}\text{Ge}$ and $^{70}\text{Ge(p2n)}$ $^{69}\text{As}$ ($T_{1/2} = 15.2$ min) $^{69}\text{Ge}$ reactions is 61 MBq(μA·h), $P_c = 20.6\%$, and the reaction yield of $^{70}\text{Ge(p, pn+2n)}$ $^{69}\text{Ge}$ is $W = 2.7$ atoms/1000 protons.

Very frequently a radionuclide can be formed through several reactions for which various stable isotopes of the irradiated element serve as target nuclei. In many cases it is possible both to identify the reaction which makes the largest contribution to the nuclide yield and to determine the yield of this reaction. The yield of the main reaction can be calculated by the formula

$$W_0 = \frac{(W - W_1 P_1 - W_2 P_2 - \ldots)}{P_0},$$

(3)

where $W$ is the total yield of all reactions (atoms/1000 protons) calculated from the measured yield of the radionuclide by means of formula (1) for $P_c = 100\%$; $W_1$, $W_2$ ... are the yields of other reactions making a lesser contribution to the radionuclide's yield, in atoms/1000 protons; $P_1$, $P_2$ ... is the concentration of target isotopes for other reactions, in %; and $P_0$ is the target isotope concentration for the main reaction, in %.
Experimental values of $W_1$, $W_2$, ... are unknown, so in their place we use average values of known yields for reactions of this type in the appropriate nuclear mass range. Let us consider a practical example of how to calculate the yield of the main reaction from Eq. (3). In Ref. [1] we show (Table 1) that irradiation of neodymium produces $^{143}$Nd through $^{143}$Nd (12.2%) (pn), $^{144}$Nd (23.87%) (p2n), $^{145}$Nd (8.3%) (p3n) reactions: $^{143}$Nd (12.2%) (pn), $^{144}$Nd (23.87%) (p2n), $^{145}$Nd (8.3%) (p3n) (the concentrations of $^{143}$, $^{144}$, $^{145}$Nd in natural neodymium are given in brackets); the $^{143}$Pm yield is 0.59 MBq/(μA·h). The table below shows that in the given nuclear mass region the p2n reaction has a substantially higher yield than the pn and p3n reactions and that the $^{144}$Nd concentration is higher than that of $^{143}$Nd and $^{145}$Nd. This is why the $^{144}$Nd (p2n)$^{143}$Pm reaction makes the largest contribution to the $^{143}$Pm yield and constitutes the main reaction.

From the table we find that $W_1 = 0.19$ atoms/1000 protons from the $^{143}$Nd (pn)$^{143}$Pm reaction constitutes the mean of the pn reaction yields for the $^{139}$La, $^{154}$Sm and $^{153}$Eu nuclei. Similarly, values of the p3n reaction yields for $^{128}$Tl, $^{152}$Sm and $^{170}$Er give $W_2 = 0.32$ atoms/protons for the $^{145}$Nd (p3n)$^{143}$Pm reaction. The total yield from all reactions leading to the formation of $^{143}$Pm is $W_n = 0.554 T_{1/2} (B/100) = 0.86$ atoms/1000 protons. Now all the quantities needed to compute the yield of the $^{144}$Nd (p2n)$^{143}$Pm reaction using formula (3) have been defined: $W_o = 3.4$ atoms/1000 protons.

The main reaction yield calculated from Eq. (3) contains an error due to the uncertainty in the values of $W_1$, $W_2$, ... used. Where, however, the contribution of other reactions to the nuclide yield is small, the errors in $W_1$, $W_2$, ... do not significantly affect $W_o$. For the purposes of the present example, if we use values of $W_1$ and $W_2$ which are increased or reduced by a factor of 1.5, the yield $W_o$ will differ by only 0.1 from the quantity 3.4 while changing the values of $W_1$ and $W_2$ by a factor of 2 will change $W_o$ by 0.2.

Let us now evaluate the contribution of the pn and p3n reactions to the yield of $^{143}$Pm for $W_1 = 0.19$ atoms/1000 protons and $W_2 = 0.32$ atoms/1000 protons. From Eq. (1) we obtain:

$$B = \frac{1.81 WP}{(Z A T_{1/2})},$$  \hspace{1cm} (4)

The yield of $^{143}$Pm given by Eq. (4) for the $^{143}$Nd(pn)$^{143}$Pm reaction is $B_1 = 0.016$ MBq/(μA·h), and for the $^{145}$Nd(p3n)$^{143}$Pm reaction $B_2 = 0.018$ MBq/(μA·h); the combined yield is 0.034 MBq/(μA·h), which accounts for 5.8% of the total $^{143}$Pm yield [0.59 MBq/(μA·h)]. Consequently, when the contribution
of other reactions to the nuclide yield is approximately 6%, a change in W₁, W₂ by a factor of 1.5 or even 2 does not materially alter the main reaction yield. Obviously, when the contribution of other reactions to the radionuclide yield is greater, changes in W₁, W₂ will influence the size of the yield from the main reaction more powerfully. The following example illustrates this. Irradiating platinum produces ¹⁹⁴Au through the reactions ¹⁹⁴Pt(pn), ¹⁹⁵Pt(p₂n) and ¹⁹⁶Pt(p₃n), and the ¹⁹⁴Au yield is 78 MBq/(μA·h) (see Ref. [1], Table 1). From the table in this paper it can be seen that in this nuclear mass range the p₂n reaction yield is markedly higher than the pn and p₃n reaction yields. Let us calculate the yield of the ¹⁹⁵Pt(p₂n)¹⁹⁴Pt reactions from Eq. (3), using the data in our present table to find W₁ and W₂. We define the yield of the ¹⁹⁴Pt(pn)¹⁹⁴Au reaction as the mean yield of the pn reactions on ¹⁸⁴W, ¹⁹⁷Au and ²⁰²Hg: W₁ = 0.13 atoms/1000 protons; the yield of the ¹⁹⁶Pt(p₃n)¹⁹⁴Au reaction as the mean yield of the p₃n reaction on ¹⁸⁷Re, ²⁰²Hg, ²⁰⁴Hg and ²⁰³Tl: W₂ = 0.47 atoms/1000 protons. Given these values of W₁ and W₂, the yield of the ¹⁹⁵Pt(p₂n)¹⁹⁴Au reaction amounts to W₀ = 1.6 atoms/1000 protons. If we increase W₁ and W₂ by a factor of 1.5, we reduce W₀ by 13%; and if we reduce W₁ and W₂ by a factor of 1.5, W₀ is increased by 8%. The combined contribution of pn and p₃n reactions to the ¹⁹⁴Au yield, given these values of W₁ and W₂, is 23% of the ¹⁹⁴Au yield. Calculating the reaction yield from Eq. (3), we first determined the contribution of other reactions to the radionuclide yield for the given values of W₁, W₂; the main reaction yield was calculated in those cases where the contribution of other reactions to the radionuclide yield did not exceed approximately 25%. We may assume that with this procedure the error is W₀ due to the uncertainty in the W₁, W₂... values used does not exceed 10-15%; this is approximately equal to the error in the experimental radionuclide yields quoted in Ref. [1]. Cases where the reaction yield was calculated from Eq. (3) are identified in the table by the letters "mr" (main reaction). In the overwhelming majority of our calculations the contribution of other reactions was less than 10%. The yields obtained from Eq. (3) were a valuable supplement to the reaction yield data.
Let us consider one other method of determining reaction yields. If a nuclide is formed from two reactions whose thresholds differ markedly from one another, the yield of each reaction can in some cases be ascertained from the curve of the nuclide yield versus proton energy. Typical examples are the experimental curves of the $^{113}$Sn [2] and $^{203}$Pb [3] yields resulting from the irradiation of indium and thallium. In both cases the nuclide is formed by the pn and p3n reactions. Up to the threshold of the p3n reaction at approximately 18 MeV the nuclide is formed only via the pn reaction; when the proton energy exceeds 18 MeV we can observe a rapid increase in the nuclide yield as the p3n reaction comes into play. Extrapolating the nuclide yield curve due to the pn reaction from $E_p = 18$ MeV to $E_p = 22$ MeV, we can obtain the nuclide yield at $E_p = 22$ MeV due to the pn reaction alone. The difference between the total nuclide yield and the yield from the pn reaction gives the nuclide yield from the p3n reaction at $E_p = 22$ MeV. From the nuclide yields found in the pn and p3n reactions we can calculate the pn and p3n reaction yields using Eq. (1). The reaction yields determined in this way are identified in the table by the letters "yc" (yield curve). The $^{110}$In and $^{111}$In yield in the pn and p2n reactions was measured by irradiating enriched $^{111}$Cd and $^{112}$Cd (98% enrichment). The yields of these reactions are identified by "enr".

Certain patterns can be observed in the way reaction yields change as the atomic number of the target nucleus increases: the pn reaction yield diminishes while that of the p3n reactions increases. In the case of p2n and pnn reactions, the yield has a certain maximum value, and the pnn reaction yield changes relatively little. Careful study of the table will reveal other patterns which can be explained if we take into account the change in nuclear binding energy per nucleon, the magnitude of the reaction thresholds, the role of the Coulomb barrier and the specific reaction mechanism. It is also clear that the yield of isomer formation reactions is substantially lower, a fact which is evidently accounted for by the lower probability of levels with higher spin being populated in the final excited nucleus. The anomalously high yield of the $^{58}$Ni(p2p)$^{57}$Co reaction is of special interest here.
The reaction yield values shown in the table are of practical use in evaluating the yields of various radionuclides obtained when targets are irradiated with 22 MeV protons and protons at other energy levels. To estimate the radionuclide yield obtained with protons of different energies, we can make use of the relative behaviour of the nuclide yield curves for reactions of a particular type - \( p_n \), \( p_2n \), \( ppn \) and so on - in the nuclear mass range of interest. Normalizing the relative shape of the nuclide yield curve to a value of \( W \) (for \( E_p = 22 \text{ MeV} \)) selected on the basis of the data in the table, we obtain from the normalized curve values of \( W \) for the selected reaction type at \( E_p = 22 \text{ MeV} \). The radionuclide yield for the value obtained is then calculated by means of Eq. (4). Examples of radionuclide yield curves for various types of reactions with protons can be found in Ref. [4].

It is proposed to publish systematized data on nuclear reaction yields for 22 MeV deuterons and 44 MeV \( \alpha \)-particles.

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REFERENCES

[1] DMITRIEV, P.P., MOLIN, G.A., Voprosy atomnoj nauki i tekhniki (Problems of atomic science and technology), Nuclear constants series, No. 5(44) (1981) 43.

[2] DMITRIEV, P.P. et al., Atomnaya Ehnergiya, 39 (1975) 135.

[3] DMITRIEV, P.P. et al., Ibid; 38 (1974) 100.

[4] DMITRIEV, P.P. et al., Ibid, 20 (1966) 154; 21 (1966) 52; 22 (1967) 310; 23 (1967) 61; 23 (1968) 278-279; 26 (1969) 467; 27 (1969) 13,125,208; 28 (1970) 503; 29 (1970) 205-206; 32 (1972) 774; 34 (1973) 404-405; 35 (1973) 61; 41 (1976) 38,431; 42 (1977) 148; 46 (1979) 53; 48 (1980) 122,402; 49 (1982) 329; 52 (1982) 72; 53 (1982) 198.
Yields from reactions of various types for a thick target bombarded by 22 MeV protons

| Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons | Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons | Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons | Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons |
|----------------|----------------|------------------------------------|----------------|----------------|------------------------------------|----------------|----------------|------------------------------------|----------------|----------------|------------------------------------|
| 7Li            | 3He            | 3.4                                | 66Zn           | 66Ga           | 0.66 br                            | 110Cd          | 110Mo          | 0.33 enr*                           |
| 11B            | 11C            | 2.6                                | 69Ga           | 69Ge           | 1.4                                | 111In          | 111In          | 0.93 enr                           |
| 13C            | 12C            | 1.7                                | 72Ge           | 72As           | 2.4 br                             | 99Cd           | 99Cd           | 0.38 br                           |
| 18O            | 18F            | 3.2                                | 76As           | 76Se           | 0.76                               | 120Sb          | 120Sb          | 0.12 br                           |
| 26Mg           | 26Al           | 0.64                               | 75Kr           | 75Kr           | 0.95                               | 121Sb          | 121Sb          | 0.52                             |
| 44Ca           | 44Sc           | 1.3                                | 85Sr           | 85Sr           | 1.3                                | 122Sb          | 122Sb          | 0.31 br                           |
| 45Sc           | 45Ca           | 0.62                               | 88Sr           | 88Sr           | 2.4                                | 123Sb          | 123Sb          | 0.47 br                           |
| 48Ti           | 48V            | 2.6 br*                            | 89Y            | 89Y            | 0.70                               | 125Sb          | 125Sb          | 0.47 br                           |
| 51V            | 51Cr           | 3.1                                | 92Zr           | 92Zr           | 0.78 br                            | 126Sb          | 126Sb          | 0.47 br                           |
| 52Cr           | 52Mn           | 0.7I br                            | 93Mo           | 93Mo           | 0.048                              | 130Se          | 130Se          | 0.32                             |
| 54Cr           | 54Fe           | 1.2                                | 93Se           | 93Se           | 0.088                              | 132Sn          | 132Sn          | 0.33                             |
| 55Mn           | 55Fe           | 1.5                                | 94Cd           | 94Cd           | 1.0                                | 133Ba          | 133Ba          | 0.46                             |
| 56Fe           | 56Co           | 1.3 br                             | 103Pd          | 103Pd          | 1.0 br                             | 135Ba          | 135Ba          | 0.34                             |
| 57Fe           | 57Co           | 1.2 br                             | 107Ag          | 107Ag          | 1.0                                | 154Ba          | 154Ba          | 0.13                             |
| 60Cu           | 65Zn           | 2.6                                | 109Ag          | 109Ag          | 1.0                                | 155Ba          | 155Ba          | 0.13                             |

* The meaning of "br", "yc" and "enr" is explained in the text.
| Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons | Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons | Target nucleus | Product nucleus | Reaction yield, atoms/1000 protons |
|----------------|-----------------|-----------------------------------|----------------|----------------|-----------------------------------|----------------|----------------|-----------------------------------|
| 153Cd          | 133Ba           | 0.1I 18 br                       | 202Ta         | 201T1         | 1.7 br                           | 106Ag         | 189Br         | 0.083 yc                          |
| 165Er          | 143Ce           | 0.18 br                          | 203Hg         | 202Th         | 0.83 br                          | 107Au         | 189Sn         | 0.78 yc                           |
| 177Er          | 153Eu           | 0.08 br                          | 203Tl         | 202Bi         | 2.1 br                           | 102Pa         | 207Bi         | 0.93 br                           |
| 181Ta          | 153Sn           | 0.32 br                          | 203Bi         | 202Bi         | 1.8 br                           | 202Bi         | 202Bi         | 0.52 yc                           |
| 181W           | 153Sn           | 0.066 yc                         | 111In         | 133Ba         | 0.67 br                          | 107Bi         | 202Bi         | 0.34 yc                           |
| 193Au          | 173Lu           | 0.47 yc                          | 111In         | 133Ba         | 0.53 br                          | 107Bi         | 202Bi         | 0.21 br                           |
| 205Tl          | 203Bi           | 0.13 yc                          | 111In         | 133Ba         | 0.48 br                          | 107Bi         | 202Bi         | 0.50 yc                           |
| 48Ca           | 44Sc            | 0.47 yc                          | 111In         | 133Ba         | 0.25 br                          | 107Bi         | 202Bi         | 0.20 br                           |
| 45Sm           | 52Co            | 0.14 yc                          | 111In         | 133Ba         | 0.12 br                          | 107Bi         | 202Bi         | 0.036                             |
| 63Cu           | 67Ga            | 2.6 br                           | 111In         | 133Ba         | 0.31 br                          | 107Bi         | 202Bi         | 0.13 br                           |
| 68Zn           | 69Ga            | 1.8 I                            | 111In         | 133Ba         | 0.094                            | 107Bi         | 202Bi         | 0.017 br                          |
| 72Mo           | 73As            | 1.7 br                           | 111In         | 133Ba         | 0.0072                           | 107Bi         | 202Bi         | 0.0043 br                         |
| 87Ge           | 88Yf            | 2.1 br                           | 111In         | 133Ba         | 0.0072                           | 107Bi         | 202Bi         | 2.4 br                            |
| 92Zr           | 95Nb            | 0.16 br                          | 111In         | 133Ba         | 0.12                            | 107Bi         | 202Bi         | 0.011 br                          |
| 98Mo           | 100Cd           | 1.7 br                           | 111In         | 133Ba         | 0.064 br                         | 107Bi         | 202Bi         | 0.0034 br                         |
| 109Pd          | 114Cd           | 0.7 br                           | 111In         | 133Ba         | 0.062                            | 107Bi         | 202Bi         | 0.0034 br                         |
| 112Cd          | 119Sn           | 0.19 br                          | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.12                              |
| 118Sn          | 123I            | 1.6 br                           | 111In         | 133Ba         | 0.16                            | 107Bi         | 202Bi         | 0.064                             |
| 120Te          | 127I            | 2.8 br                           | 111In         | 133Ba         | 0.17                            | 107Bi         | 202Bi         | 0.054                             |
| 128Re          | 131Sn           | 1.9 br                           | 111In         | 133Ba         | 0.22                            | 107Bi         | 202Bi         | 0.019                             |
| 131Ba          | 132Sn           | 0.80 br                          | 111In         | 133Ba         | 0.062                            | 107Bi         | 202Bi         | 0.33 br                           |
| 135La          | 139Ce           | 1.17 br                          | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.22                              |
| 140Gd          | 140Pr           | 3.4 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.037                             |
| 144Sm          | 147Eu           | 2.2 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.057                             |
| 148La          | 151Eu           | 1.2 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.019                             |
| 166Er          | 157Eu           | 1.7 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.33 br                           |
| 177Lu          | 164Tm           | 2.5 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.10 br                           |
| 177Lu          | 164Tm           | 0.7 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.011                             |
| 182Ta          | 165Tm           | 1.5 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.011                             |
| 182Re          | 165Tm           | 1.8 br                           | 111In         | 133Ba         | 0.063                            | 107Bi         | 202Bi         | 0.011                             |
| 183W           | 165Tm           | 0.52 yc                          | 111In         | 133Ba         | 0.064 yc                         | 107Bi         | 202Bi         | 0.017                             |
| 185W           | 165Tm           | 0.62 yc                          | 111In         | 133Ba         | 0.044 yc                         | 107Bi         | 202Bi         | 0.21                              |
| 197Pt          | 173Lu           | 1.5 br                           | 111In         | 133Ba         | 0.044 yc                         | 107Bi         | 202Bi         | 0.037                             |
| 197Pt          | 173Lu           | 1.5 br                           | 111In         | 133Ba         | 0.044 yc                         | 107Bi         | 202Bi         | 0.21                              |
| 205Tl          | 207Bi           | 0.81 br                          | 111In         | 133Ba         | 0.044 yc                         | 107Bi         | 202Bi         | 0.037                             |
AN INTEGRATED HARDWARE/SOFTWARE CONFIGURATION FOR THE EVALUATION OF NUCLEAR DATA

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The large volume of data which an evaluator has to process in order to obtain recommended constants is the determining factor in evaluation work. Reference [1] describes a data bank set up to make the evaluator's work more efficient. Today, the use of a data bank without man-machine interaction by means of computer graphics would be unthinkable. Man-machine interaction in this context means not only the hardware and software required for data storage and retrieval in a nuclear data bank but also a simple and reliable means of entering the data into the bank from a range of data carriers. Experience shows that organizing the collection of data presented in graphic and tabular form is a vital aspect of this task.

In order to carry out his evaluation, the physicist must have the following technical resources at his disposal: (1) digitized versions of the graphics with subsequent recording of the digitized data onto a machine carrier or direct to the computer; (2) visualization of the input on a display unit; (3) direct access to a central data bank from remote terminals; and (4) buffered input and review of all data entered. The software must include input and edit programs for all data and ancillary information entered and also format modification programs. The article reviews a hardware/software configuration designed along modular lines for tasks involving graphically presented information.

Hardware

The basic set of hardware modules required to process graphical data includes (Fig. 1):

- An acoustic device for digitizing graphics, a so-called acoustic pen (AP);
- A device to display graphics data on the CRT screen (GDD);
- Magnetic tape storage;
- An alpha-numeric display unit with an attachment for graphics generation;
- A wide-range, high-speed recorder;
- A type "Ehlektronika DZ-28" microcomputer; and
- A PL-150 perforator.

All these modules are combined in an integrated system by means of a shared data line (SDL) of type MEhK 625.1 [2]. The SDL MEhK 625.1 system was chosen as the main line, firstly because it is relatively simple, secondly because it can be used as the basis for simple, controllerless systems of the "transmitter only-receiver only" type, and thirdly because this standard is coming into ever wider use at the present time.

**Graphics digitizer. Acoustic pen (AP)**

The acoustic digitizer for graphical information consists of a plotting screen on which graphics can be set out or projected; a spark pen which generates a sound-wave at the point of contact; and a control unit and an interface unit. When the spark discharger at the tip of the pen is activated, the counters begin to record X and Y impulses generated by the master clock. The clock ceases to function when the sound-wave reaches the linear microphones mounted round the edges of the plotting screen. In order to prevent low-frequency sound from being superimposed on the desired signal, electrical oscillations from the microphones are fed to amplifiers with a restricted transmission band.

After the fast discriminators have been tripped by the steep leading edge of the oscillations (this also stops the master clock), numbers proportional to the distances from the discharger to the microphones are formed in the X and Y counters. The co-ordinate pair thus obtained is transformed into a sequence of octal coded messages in the KOI-7 codes and is then transmitted to the SDL by means of 8 bytes. The codes for the symbols "line feed" and "carriage return" are transmitted at the end of the message. In addition, command data such as "write to internal memory", "delete period", "tape marker write-in" etc. can be transmitted via the SDL to the receiving terminal. There are control buttons for this purpose located on the front panel of the acoustic pen control unit. The plotter has a working area of 350 mm², the
The number of dots is 1024 x 1024, and instrument error is ± 0.35 mm. The AP interface is fully consistent with the standard [2] and can carry out an interrupt function in addition to the data transmission function.

**Device for graphics display on the CRT screen**

This device provides visual control over the digitization of graphics data. In essence, the device consists of two digital-analog converters together with a unit linking them to the SDL. Iterative sequences of coded messages serve as input data. These are generated by the line transmitter and carry information on the co-ordinates of the illuminated dots. Any oscillographic apparatus may be used to depict the output analog signals, provided its horizontal deflection system can be governed by an external clock and the brightness of its beam can be modulated by an external signal. As the data enter the system incorporating a graphics display unit, the system controller can correct the data array appearing on the CRT and thereby alter the display dynamically.

**Magnetic tape unit**

The large flow of graphics data entered into the nuclear data bank and the relatively slow speed of digitization impose a specific mode of processing which may make it preferable to hold the data initially on magnetic or punched tape rather than entering them directly into the computer.

The ES-9002 standard data preparation tape drive (DPTD) was used for magnetic tape storage. This tape drive is fitted with a unit linking it to the SDC. In addition, the DPTD has been further modernized, can accommodate records of any length and can operate with the KOI-7 codes which are standard with this system. In relation to the main line, the ES-9002 can be both a receiver and a transmitter, which means that, in addition to output to a machine carrier, the reverse operation, i.e. entering data into the system, can be carried out. The structure of the SDL link unit is such that the ES-9002 DPTD's buffer can regenerate its contents onto the main line if so instructed by the controller, i.e. can thereby function as a transmitter of information to the DRGT for illumination on the CRT screen.
Connecting the graphics processing hardware to the computer

A display plugged into the computer in the standard manner and fitted with an SDL linkage unit is used to link up the system. The linkage unit comprises a multiplexer and interface circuits (Fig. 2). The codes of the symbols are transmitted to the display and passed on to the computer either from the keyboard or from the SDC depending on the position of the toggle switch on the keyboard. This solution makes it possible to dispense with special systems programs and to use the standard driver for the particular terminal.

Such visual display terminals equipped with a graphics module which allows the display of both alpha-numeric and graphical information have a much wider range of functions and a greater degree of perfection from the standpoint of graphics processing. The graphics module is constructed on a principle that is in wide use at the present time - the contents of two memories (an alpha-numeric and a graphics memory) are regenerated on the screen simultaneously. By means of a special control sequence the display switches to a graphics mode and treats the subsequent flow of data from the computer as command information with which to generate an image. As a rule, either the co-ordinates of the individual dots on the screen are transmitted or the starting and ending co-ordinates of the vectors which are to be illuminated. The microprocessor controlling the display carries out the necessary modifications and fills the graphics memory. Another control sequence is used for translation to the normal alpha-numeric mode. The working area of the graphics raster is 272 x 240 dots. The KDE 810A860 display unit serves as a source.

Special subprograms for use with the SM3 and SM4 operating systems have been developed for the benefit of users who wish to program their own graphics in high-level languages such as FORTRAN.

Available system configurations

The use of trunk-linked modules in designing the graphics processing system made it possible to set up mutually complementary systems at minimal cost. In this way, data could be entered into a nuclear physical constants base and used in various ways (off-line, on-line etc.). Figures 3 and 4 illustrate some of the more frequently encountered configurations. The systems depicted in Figs 3, 4(a) and 4(b) are the simplest and can be utilized where a computer with appropriate resources is not available in the immediate vicinity.
In cases where the quality of the input is of special importance, and primary processing is best carried out off-line (i.e. is not possible on the central computer), the configuration depicted in Fig. 3(c) should be used. For primary processing of input data, a type "Ehlektronika DZ-28" microcomputer connected to the SDL can be used. With this set-up, data obtained with the systems illustrated in Figs 3(a) and (b) can undergo primary processing. Figure 4 depicts the most suitable configuration in cases where there is direct access to the computer. With this system graphics data can be input to the computer direct without intermediate carriers, and processed on-line.

The configurations depicted in Figs 4(b) and 5 are used to enter experimental data to the base direct from the recorder. Figure 4(b) illustrates a system which enables data from the "Fialka-4M" recorder [3] to be entered to the computer via a video terminal. Two k-bytes of data can be transmitted in the space of 2 minutes to 10-15 seconds, depending on how the video terminal is connected to the computer. Figure 5 depicts an off-line system which enables the internal storage of all (max. 31) recorders to be written into standard files held on magnetic tape for subsequent computer processing. The algorithm of the control program for the system controller (an "Ehlektronika DZ-28" microcomputer) contains a number of effective design features to prevent data corruption or loss. In addition, data can also be recorded on a cassette deck which is part of the "Ehlektronika DZ-28" standard equipment.

Software resources

The software resources of the graphics data processing system comprise the following: (1) primary processing; (2) search and recording of data in the nuclear data bank; and (3) processing of data in the base. The latter two sets of programs are described in Ref. [1]. The primary processing programs include: programs to enter data with the acoustic pen and record them in the file; edit programs for these data; programs to transfer the co-ordinates to a physical system and correct various non-linear distortions along the axes of these co-ordinates; and so on. The introduction of functional keys by means of which all further processing of primary graphics data can be largely automated is a feature which makes the system particularly convenient for the evaluator. The functional keys appear on the plotting screen as squares of 1 x 1 cm, positioned somewhat to the side of the basic working image. By pressing the tip of the pen inside the square, the code of the functional key is transmitted to the output file. The following functional keys are available:
BEG - open bracket (.
END - close bracket ).
LNX - open bracket for an array of dots marking the X-axis.
LNY - ditto for the Y-axis.
SCX - open bracket for an array of dots marking a scale on the
      X-axis.
SCY - ditto for the Y-axis.
LIN - type of scale (linear).
LOG - type of scale (logarithmic).
NLN - type of scale (non-linear).
ERX - defines the type of dot which precedes this key as the
      deviation (yc) along the X-axis of the dot preceding the
      dot "yc".
ERY - ditto for a deviation along the Y-axis.
FLG - marker key. The presence of this key permits the index of
      all subsequent dots to be increased by one.
CLF - flag (index) clearance.
NGR - key to reserve space for input of tabular data from a table.
NSC - the presence of this code indicates that scale data are
      unavailable for one of the axes used instead of SCX or SCY.
LAC - key to enter the amplitude calibration lines.
REP - marker key. Indicates that the preceding dot is for
      reference only.

The operations required to digitize total initial data can be described
in brief as follows using Naur-Bekus's terminology:

<Operation> ::= BEG <all graphics> END
<All graphics> ::= = <graphic>
<all graphics> / <graphic> NGR/
<all graphics> / NGR ... NGR
Thus, processing the graphic information entered in accordance with the above protocol can be almost entirely automated, reducing manual operations to a minimum. Nuclear data evaluated by this system can be recorded and stored in an exchangeable format of the EXFOR type [4].
REFERENCES

[1] Newsletter of the Nea Data Bank. Nuclear Energy Agency, N 25, January, 1981.

[2] IEEE-STD 488-1975. IEEE STANDARD. Interface for Programmable Instrumentation (Hewlett-Packard Co., April 4, 1975), GOST 23131-80. Aggregated facilities for information and measurement systems. Specifications for cross-section parameters in radial and chain connection of data collection and formatting systems.

[3] GORYUSHKIN, S.I., OBSHIVALOV, V.A., ROSTOVTSEV, A.A., SHEIN, Yu.V., A device with automatic selection of the range of measurement for recording the shape of simplex signals. Pribory i tekhnika eksperimenta (Experimental apparatus and techniques) No. 3 (1977) 121.

[4] DUNFORD, C.L., BERLAND, R.F., HUBNER, R.F., GREASY, R.J., Score II an Interactive Neutron Evaluation System (Atomics International, Canaga Park, California, March 1, 1969). AI-AEC-12757.

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Fig. 1. Basic configuration of hardware modules: 1 - terminal channel; 2 - visual display unit with graphics option (attachment); 3 - F-4M recorder; 4 - "Ehlektronika DZ-28" microcomputer; 5 - modernized type ES-9002 DPTD.

Fig. 2. Unit linking the KDE 81GA860 to the SDL: 1 - keyboard; 2 - multiplexer; 3 - interface; 4 - SDL linkage unit; 5 - display; 6 - terminal channel.
Fig. 3. Off-line graphics digitizing system with (a) paper tape; (b) magnetic tape; and (c) with visual control and edit option: 1 - PL-150 perforator; 2 - modernized type ES-9002 STPTD; 3 - "Ehletronika DZ-28" microcomputer.
Fig. 4. System for entering graphical (a) and experimental (b) information to the computer with processing and visual control occurring simultaneously: 1 - terminal channel; 2 - type KDE 810A860 video terminal; 3 - F-4M recorder.

Fig. 5. Off-line system by which the contents of the memories of all recorders can be written into standard files: 1 - "Ehlektronika DZ-28" microcomputer; 2 - F-4M recorder; 3 - modernized type ES-9002 SDPTD.