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PROBLEMS OF RADIATION SAFETY IN NUCLEAR
POWER STATIONS CONTAINING VVÉR-440 REACTORS

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Since the assembly and initiation of the first standard-production nuclear power stations containing VVÉR-440 reactors, the Novovoronezh station (third and fourth units) has been operating for more than four years and the Kol'sk station (first and second units) for about three [1]. Existing data indicate that the radiation safety of both personnel and local population has been reliably ensured.

The efficiency of the biological shielding against the n and γ radiation from the active zone of the reactor and the pipelines containing the circulating coolant was studied at the time of the physical initiation of the units and again after power production had begun.

In the period of physical initiation, access was available to the equipment of the first circuit and to the space immediately surrounding the reactor, so facilitating the study of protective measures in these zones.

After a certain period of operation at nominal power, information was collected not only regarding the radiation from the activated atoms of the water coolant but also regarding the active fission products and corrosion which had accumulated in the water and on the equipment of the first circuit.
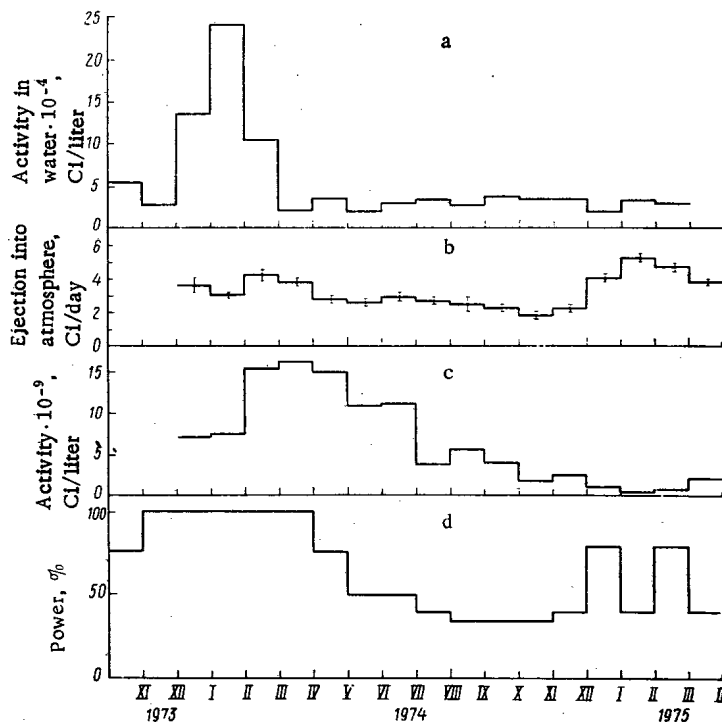
Fig. 1. Total activity of ^{85m}Kr, ^{88}Kr, ^{133}Xe, ^{115}Xe in the water of the first circuit (a); ejection of radioactive gases into the atmosphere (b); activity of the gases in the air of the technological rooms (c); reactor power (d).

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TABLE 1. Calculated and Actual Content of Various Fission and Corrosion Products in the Cooland and Deposits of the VVÉR-440 Reactor

Radionuclide	Activity of coolant, Ci/liter		Activity of deposits	
	calc.	actual*	Ci/cm^2	contribution to dose, %
^{133}Xe	$9,3 \cdot 10^{-2}$	$2,4 \cdot 10^{-3}$	—	—
^{135}Xe	$1,2 \cdot 10^{-2}$	$3,0 \cdot 10^{-4}$	—	—
^{85m}Kr	$2,0 \cdot 10^{-4}$	$6,1 \cdot 10^{-5}$	—	—
^{88}Kr	$2,1 \cdot 10^{-3}$	$8,1 \cdot 10^{-5}$	—	—
^{3}H †	$8,2 \cdot 10^{-5}$ $1,0 \cdot 10^{-4}$	$2 \cdot 10^{-5}$— $-3 \cdot 10^{-6}$	—	—
^{139}Ba	$5,7 \cdot 10^{-4}$	$1,5 \cdot 10^{-5}$	—	—
^{140}La	$6,3 \cdot 10^{-6}$	—	$3,8 \cdot 10^{-8}$	4,1
^{131}I	$1,7 \cdot 10^{-3}$	$4,6 \cdot 10^{-5}$	$1,3 \cdot 10^{-8}$	0,2
^{133}I	$4,3 \cdot 10^{-3}$	$1,3 \cdot 10^{-4}$	—	—
^{135}I	$3,3 \cdot 10^{-3}$	$1,2 \cdot 10^{-4}$	—	—
^{91}Sr	$1,7 \cdot 10^{-4}$	$2,5 \cdot 10^{-6}$	—	—
^{92}Sr	$3,0 \cdot 10^{-4}$	$1,2 \cdot 10^{-5}$	—	—
^{60}Co	$4,0 \cdot 10^{-6}$	$1 \cdot 10^{-8}$	$1,4 \cdot 10^{-7}$	19,0
^{58}Co	—	$8 \cdot 10^{-9}$	$3,0 \cdot 10^{-7}$	17,0
^{54}Mn	—	$2 \cdot 10^{-8}$	$2,5 \cdot 10^{-7}$	12,5
^{59}Fe	$7,7 \cdot 10^{-5}$	$1 \cdot 10^{-8}$	$4,5 \cdot 10^{-8}$	2,9
^{65}Zn	—	—	$9,0 \cdot 10^{-8}$	2,8
^{95}Zr	$1,4 \cdot 10^{-5}$	$5 \cdot 10^{-9}$	$3,6 \cdot 10^{-8}$	1,4
^{100m}Ag	—	$6 \cdot 10^{-8}$	$2,6 \cdot 10^{-7}$	38,5
^{16}N	$1,2 \cdot 10^{-1}$	$\sim 1,0 \cdot 10^{-1}$	—	—
^{24}Na	—	$5 \cdot 10^{-5}$	—	—

*For a nonhermeticity of the fuels elements amounting to 0.07 of the permitted value.

†The first value of the ^{3}H activity arises from triple uranium fission, the second from the reaction ^{10}B(n, 2d) ^{3}H.

These investigations showed that the level of penetrating radiations encountered during the operation of the reactor at nominal power never exceeded 28, 2.8, and 1.4 mrem/h, as respectively specified by health rules OSP-72 [2] for unserviced and partly serviced rooms and rooms constantly occupied by personnel. In some cases the figures were well below the design values. Individual slight defects in the biological shielding were discovered, but these did not result in any serious irradiation of the personnel, being of a local character and occurring in unserviced zones (these defects resulted from deficiencies in the casting of the concrete and the filling of the serpentinite during the construction of the nuclear power station, or from minor constructional shortcomings).

Thus, the weight and size of the biological shielding of the VVÉR-440 might reasonably be reduced, subject to improvements in construction quality, in conformity with the low annual dose of irradiation received by the personnel, on the average not exceeding 5% of the maximum permissible dose. The personnel in fact received about 70% of this dose during the recharging of the nuclear fuel and preventive maintenance in the shut-down reactor on account of the γ radiation arising from radioactive deposits of corrosion products (^{110m}Ag, $^{60, 58}$Co, ^{54}Mn, and others) in the equipment (Table 1).

The radiation characteristics inside and outside the nuclear power station may be greatly affected by the radioactivity of long-lived nuclides accumulating in and migrating from the coolant during service. The level of activity is determined by the number of fuel elements with damaged cans, the degree of damage, and also the activation of the corrosion products circulating through the active zone. The VVÉR-440 is only allowed to be used if no more than 1% of the fuel elements are lacking in gas-tightness and no more than 0.1% allow contact between the coolant and the fuel.

The calculated composition (Table 1) of gaseous and volatile fission products corresponds to this number of defective fuel elements in the active zone of the VVÉR-440. Table 1 also gives the measured specific activity of several corrosion products in the water and in the deposits on the surface of the equipment in the first circuit; the dose contribution of the γ radiation of the deposits is also indicated.

Considering the major contribution of ^{110m}Ag and other corrosion products to the dose, technical measures are now being developed and initiated for removing these products as quickly as possible from the coolant and reducing their proportion in the deposits.

TABLE 2. Radionuclides in the Air of the Technological Rooms, and Their Ejection into the Atmosphere of the Kol'sk Nuclear Power Station

Radionuclide	Activity in air, Ci/liter			
	technological rooms	before entering the gas filters	after passing the filters	ventilation pipe
^{131}I	$2,7 \cdot 10^{-14}$	$4,35 \cdot 10^{-12}$ * $1,21 \cdot 10^{-10}$ †	0 0	$2,5 \cdot 10^{15}$*
^{133}I	$3,2 \cdot 10^{-14}$	$42,2 \cdot 10^{-11}$ * $4,6 \cdot 10^{-10}$ †	0 0	—
^{133}Xe	$7,6 \cdot 10^{-10}$	$2,2 \cdot 10^{-6}$	$6,3 \cdot 10^{-8}$	$3,5 \cdot 10^{-11}$
^{135}Xe	$1,4 \cdot 10^{-10}$	$9,0 \cdot 10^{-7}$	$1,4 \cdot 10^{-8}$	$7,0 \cdot 10^{-12}$

*In aerosol form.
†In volatile form.

TABLE 3. Calculated Average Annual Doses of External and Internal Irradiation Experienced by the Population in the Region of a Nuclear Power Station Containing a VVÉR-440 Reactor, mrem/yr

Radionuclide	External irradiation		Internal irradiation	
	projected dose	actual dose	projected dose	actual dose
^{85m}Kr	$3,5 \cdot 10^{-3}$	$9,0 \cdot 10^{-6}$	—	—
^{87}Kr	$6,7 \cdot 10^{-2}$	$1,6 \cdot 10^{-4}$	—	—
^{88}Kr	$3,2 \cdot 10^{-1}$	$8,0 \cdot 10^{-4}$	—	—
^{3}H	—	—	$1,5 \cdot 10^{-3}$	$4,0 \cdot 10^{-6}$
^{133}Xe	$3,1 \cdot 10^{-1}$	$7,8 \cdot 10^{-4}$	—	—
^{135}Xe	$3,8 \cdot 10^{-1}$	$9,5 \cdot 10^{-4}$	—	—
^{131}I	$1,0 \cdot 10^{-6}$	—	$23,0$	$6,0 \cdot 10^{-2}$
^{133}I	$4,0 \cdot 10^{-6}$	—	$2,2$	$5,5 \cdot 10^{-3}$
^{135}I	$1,0 \cdot 10^{-7}$	—	$0,2$	$5,5 \cdot 10^{-4}$
Total	$\sim 1,0$	$\sim 3 \cdot 10^{-3}$	$\sim 294,2$	$\sim 6,6 \cdot 10^{-2}$

Note. The projected dose refers to the outflow into the atmosphere when operating at nominal power with a limiting amount of fission products in the coolant (up to 0.1 Ci/liter) and the permitted disorganized leakage of coolant (up to 200 liters/h). The actual value is obtained for a disorganized leakage of 5 liters/h and a coolant activity of 0.01 Ci/liter, i.e., for 0.1 of the permissible number of fuel elements with damaged cans. If the coolant leakage is increased to 50 liters/h while the activity reaches the limiting value the actual doses will increase by a factor of 100 times.

We see from Table 1 that the calculated activity (for a nominal power at the instant of withdrawing the sample) for the gaseous fission products reaches 0.1 Ci/liter and for iodine isotopes 0.01 Ci/liter. Corresponding to this, an activity of ~0.01 Ci/liter attributable to nongaseous products occurs in the residue of the sample measured 2 h after selecting the latter. These values may vary by a factor of several times, depending on the level of reactor power, the rate at which the water passes to the purification system, the efficiency of the purifying filters, the intensities of the organized and disorganized water flows, the rate at which fresh water is supplied, and so forth.

The true values of the activities of the isotopes ^{131}I, ^{133}I, ^{135}I, ^{133,} ^{135}Xe, ^{85m}Kr, ^{88}Kr, and ^{91,} ^{92}Sr (at the end of the first campaign of one of the active units of a nuclear power station containing a VVÉR-440 reactor) enable us to estimate the number of leaking fuel elements in the active zone and the number with more serious can damage. The estimate is based on the generally accepted interpretation of the mechanism underlying the passage of fission products from damaged fuel elements into the coolant [3]. According to these data the can defects are 10-20 times lower than the acceptable design value.

The ejection of radionuclides from the coolant into the technological rooms and then into the external ambient is illustrated in Fig. 1. These curves characterize the first unit of the Kol'sk nuclear power station from approximately the instant of starting in 1973 to the first planned fuel recharging in 1975. We see by comparing the curves that not only the proportion of radionuclides in the coolant but also, quite clearly, the level of its disorganized (and organized) flows and the efficiency of gas purification influence the ejection of the gases into the atmosphere. For this nuclear power station the 1974-1975 average gas ejection was ~3-5 Ci/day, which indicates a fairly good hermeticity of the technological equipment containing the coolant and the satisfactory operation of the gas-trapping carbon filters.

Depending on the state of the technological equipment in the individual units, the disorganized leaks of coolant lie at a level of 5-50 liters/h, much less than the permissible value of 200 liters/h (for the Kol'sk nuclear power station the leaks average no more than 5 liters/h).

The concentration of the main radiation-hazardous isotopes in the air of the technological rooms, the technological gas-purification conduits, and the effluent pipe of the Kol'sk nuclear power station is given in Table 2. In the serviced rooms the concentration is below the sensitivity limit of the dosimetric apparatus employed.

It follows from Table 2 that the gas-purification coefficient of the filters working in the specified techno-logical mode is ~60 relative to the sum of the xenon isotopes (or relative to the sum of all the gaseous nuclides determined by other measurements not less than 130). The ten-times excess of the volatile form of iodine over the aerosol (in the technological air flows) indicates the comparatively greater danger of the volatile forms of iodine.

The radiation conditions in the external ambient of the nuclear power stations is monitored over distances up to 45-60 km. For such a low rate of outflow into the atmosphere (3-30 Ci/day with respect to the radioac-tive gases and less than 10^{-4} Ci/day with respect to iodine) it is hard to detect the effects of such nuclear power stations on the environment [4].

Measurements show that the density of the radioactive fallout in the neighborhood of the nuclear power station, the concentration of the β-active aerosols [$3 \cdot 10^{-5}$ Ci/(cm$^2 \cdot$ day) and $6 \cdot 10^{-17}$ Ci/liter on average per year referred to the total β activity], and their radionuclide composition correspond to the global fallout. The gamma background is no greater than the average over the whole country and lies in the range 7-11 μR/h, not falling at all on moving away from the power station.

No increment over the background level is found in the proportions of any of the radionuclides in the soil, plants, or cereals, nor (and this is especially important) in iodine isotopes in the milk of cows during the pas-ture period. In the water of open catchments (lakes and rivers) the total β activity is also no greater than the background value of $\sim 10^{11}$ Ci/liter.

Thus, estimates as to the dose of irradiation experienced by people living near the power station may be validly obtained by calculations based on the effluent into the ambient, rather than by direct measurements on the site.

Estimates made by the method set out in [5] are presented in Table 3 for the European part of the Soviet Union. The power-station effluent pipe was taken as being 100 m high, the wind velocity 3-4 m/sec, and the elongation of the wind rose (directional diagram) 1 : 2.

The projected dose representing the internal irradiation of the human thyroid gland with iodine isotopes is calculated on the assumption that milk from the zone of action of the effluent is only used for one six-month period (while the cows are in pasture). With regard to ^{131}I the projected dose is obtained on the assumption of an outflow of $7.0 \cdot 10^{-3}$ Ci/day (in aerosol and volatile form), for a rate of deposition from the atmosphere equal to 10^{-2} m/sec. If the activity does not pass through the food chain, the contribution of internal irradia-tion may be neglected (for example, in the case of the Kol'sk nuclear power station).

On the whole, the projected and actual external and internal irradiation doses of the population in the region of nuclear power stations with VVÉR-440 reactors do not exceed the permitted values; the external ir-radiation may amount to only 1-2% of the natural background.

The real effect of the effluent from nuclear power stations with VVÉR-440 reactors on the population is even less than that indicated, since it is difficult to allow for the following factors: the relative times spent by the population in the open and in their dwelling houses, which screen the dose rate and reduce it by a factor of several times, the type of food eaten by the people, the age composition, and so on. A four-year period of ex-perience in the use of the first standard units of nuclear power stations with VVÉR-440 reactors shows these to be quite reliable, being characterized by reasonably advanced working characteristics from the point of view of ensuring radiation safety for the service personnel and local inhabitants.

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MONITORING THE REACTIVITY OF EXTREMELY

SUBCRITICAL REACTORS BY MEANS OF REACTIVITY METERS,

CORRECTIONS BEING MADE TO THE ANALOG OF THE SOURCE

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One of the most important problems in nuclear technology is that of monitoring the degree of subcriticality of reactors and critical assemblies; this includes the continuous monitoring of the reactor as it approaches the critical state (assembly of critical mass), monitoring the changes taking place in the reactivity relative to the critical state of the reactor (slight or brief changes in reactivity), and continuous monitoring of the behavior of the reactor in the subcritical state after the emergency protective systems have operated.

In these cases problems of nuclear safety may be effectively solved by creating a device capable of monitoring the reactor in the real time scale by reference to the basic parameter, the reactivity. Such devices include analog reactivity meters based on the principle of analyzing the power prehistory [1-4] of the system.

Within the framework of the local kinetic characteristics of the reactor under consideration, the reactivity meter analyzes the behavior of the neutron flux and produces a signal proportional to the reactivity. The high sensitivity of this instrument toward reactivity indicates that, in principle, it may be successfully applied to the supercritical, critical, and subcritical operating conditions of the reactor.

However, existing reactivity meters have a dynamic range of no more than three decades with respect to power input (corresponding to the continuous calculation of reactivity, while preserving the initial conditions intact) [1-3]; this is insufficient for the long-term monitoring of reactivity in the case of severe subcriticality. Reactivity meters have proved extremely successful in monitoring variations in the reactivity of reactors relative to the critical state, in which the reactivities introduced are slight or only last for a brief period, i.e., in which the changes in input currents are no greater than two or three orders of magnitude.

Recently, further developments have occurred, leading to the appearance of the first experimental models of reactivity meters with a dynamic range extending to five orders of magnitude [4]. Such instruments have greatly extended the monitoring period of the extremely subcritical state (e.g., that which applies after the emergence safety devices have operated), but the period is still quite limited, being governed by the instant at which the neutron flux becomes comparable with the power of the neutron source, which may be of the Po— Be or any other type, as well as neutrons from the (α, n) and (γ, n) reactions. For calculating the initial conditions such devices require that the reactor should have passed into the critical state before starting the measurements.

If, however, an appropriate correction may be made for the source component S in the simulating part of the reactivity meter, then quite apart from monitoring a reactor deviating slightly from the critical state such an instrument will enable us to execute continuous monitoring during the actual assembly of the critical mass (i.e., on approaching the critical state) and also long-term continuous monitoring in the subcritical state, for example, after the safety devices have been set in operation.

It is well known that in a subcritical reactor with a steady-state neutron flux the power is proportional to the intensity of the source, and the reactivity is calculated from the relation

$$\rho = -C\,(S/N), \tag{1}$$

where $C = l/\beta$; $N = \text{const}$.

Thus, in order to simulate the source analog in the instrument correctly we must know the subcriticality of the reactor.

In this paper we shall describe three methods, not requiring any additional pieces of apparatus, which will enable us to determine the magnitude of the source components for cases involving reactivity meters.

Method of the "Shooting Source". Some time after reaching subcriticality, an equilibrium state determined by the following system of equations is established in the reactor [5]:

$$\frac{dN}{dt} = -\frac{1-(1-\beta)K_{eff}}{l}N + \sum_{i=1}^{6}\lambda_i C_i + S;$$

$$\frac{dC_i}{dt} = \frac{K\beta_i}{l}N - \lambda_i C_i. \tag{2}$$

In order to calculate the reactivity in this case we are required to realize an equation constituting the solution of (2):

$$\rho = \frac{l}{N\beta}\left(\frac{dN}{dt} + \sum_{i=1}^{6}\frac{dC_i}{dt} - S\right) = \rho_0, \tag{3}$$

which contains the unknown source S.

It may be shown that a criterion for the correct choice of bias voltage in the solving part of the reactivity meter is the condition that $\partial\rho_s/\partial t = 0$ after the "shooting" of the source.

Let us accordingly consider that the source analog is chosen in an arbitrary manner, and that in general $S \neq S_S$; the mismatch between the actual subcriticality and the readings of the instrument will then be given by the equation

$$\rho_0 - \rho_s(t) = \frac{l}{\beta N(t)}(S - S_s), \tag{4}$$

which is inconstant in time on account of the fall taking place in N(t). After differentiating Eq. (4) we obtain

$$\frac{\partial\rho_s}{\partial t} = \frac{l}{\beta N^2}\frac{\partial N}{\partial t}(S - S_s), \tag{5}$$

from which we see that equating the rate of change of the instrument readings to zero means achieving equality between the simulated and real sources, since in Eq. (5) $N \neq 0$ and $\partial N/\partial t \neq 0$.

Measurements in a uranium — graphite test-bed showed that, for $S = 10^6$ neutrons/sec, the subcriticality could be measured from zero to $-8.4\beta_{eff}$ (the total efficiency of the control-and-safety rods) to an error of 5% without first having to pass into the critical state. The error is due to an apparatus effect which may subsequently be reduced. As sensor for the measurements we used an SNM-18 counter. The counter and the reactivity meter are connected to a pulse tract, a smoothing device, and a linear intensity meter.

In order to measure deeper subcriticalities and increase the measuring accuracy it is essential to increase the power of the "shooting" source and to improve the pulse apparatus, or else to use highly efficient fission chambers. The use of fission chambers extends the range of applicability of the method, since these chambers are able to work at high temperatures (up to 400°C) and subject to a strong γ background, which enables this method to be used for energy-stressed reactors in the subcritical state. However, a powerful "shooting" source is required owing to the strong neutron background due to (γ, n) reactions.

Method of the Injection of an Unknown Absorbent. Let us assume that a neutron source of intensity S exists in the subcritical reactor. The rapid introduction of an unknown absorbent into the subcritical reactor will have the effect that a transient process described by the system of equations (2) will begin. In order to choose the value of the source component after introducing the absorbent the $\partial\rho_s/\partial t = 0$ method may be employed.

In order to increase the accuracy of the measurements (if the efficiency of the unknown absorbent is very great) the neutron flux is first reduced at a rate faster than that which would correspond to $\partial\rho_s/\partial t = 0$ in the instrument. This operation may be repeated several times, and the value of S_S may be established by the method of successive approximations. After this the instrument will indicate the total reactivity of the system.

Thus this procedure enables us to measure the subcriticality of the reactor without first bringing it into the critical state, and also at the same time to calibrate the unknown absorbent.

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<u>Method of Introducing a Calibrated Absorbent.</u> This procedure is based on the choice of a bias voltage simulating the source in such a way that the reactivity determined by the instrument corresponds to the known efficiency of the absorbent so introduced.

We see in fact from Eq. (4) that for a steady-state flux, i.e., for N(t) =const, the difference $\rho - \rho_s$ is proportional to the difference between the intensities of the real and simulated sources. Hence by choosing the reactivity bias voltage so as to be equal to the efficiency of the calibrated absorbent introduced in this way we may achieve equality between the real and simulated sources. The distinguishing feature of this method is the possibility of using it with reactivity meters having a dynamic range of two orders of magnitude. The accuracy of the subcriticality measurements here depends on how accurately we know the efficiency of the calibrated source, on the validity of the corrections made for interference, and on the extent to which the lifetime of the neutrons depends on K_{eff}.

This method was tested in a uranium — graphite installation. The subcritical reactivity was varied from 0 to $-8.4\beta_{eff}$. The error in determining the reactivity was no greater than 5-10% and was attributable to the interference of the rods. As sensor we used either an SNM-18 counter or a KNK-56 chamber.

In order to increase the accuracy of the measurements when using this method it is desirable to use powerful neutron sources. The more powerful the source in the subcritical reactor, the higher is the accuracy of the method, i.e., the introduction of a calibrated absorbent enables us to measure subcriticality in subcritical energy-stressed reactors with a substantial neutron background or in the presence of a powerful intensifying source.

CONCLUSIONS

The advantage of the instrumental methods here described lies in the fact that no additional apparatus is required for determining subcriticality; it is sufficient simply to use an analog reactivity meter allowing the source component in the solving part to be varied. By using these methods we may determine the bias voltage (constituting the analog of a steady-state source in the reactor) in the subcritical state. If the intensity of the source does not alter very sharply over the period of measurement (5-10 min), the introduction of the source function into the instrument is an "on-off" operation, after which the reactivity meter is able to monitor any changes in K_{eff} continuously without first bringing the reactor into the critical state. This offers the possibility of making quite accurate measurements to ensure nuclear safety. In cases in which it is required to secure especially accurate and reliable results, these measurements may be repeated as from the critical state as part of a total monitoring operation. The safety of operations requiring passage into the critical state is also increased, since the passage into the critical state is monitored completely.

Instrumental methods of measuring subcriticality are universal; they embrace not only critical assemblies but also any energy-stressed reactors in the subcritical state, and enable us to measure temperature effects, effects of poisoning, and so on, i.e., effects characteristic of energy-stressed reactors. In addition to this, these methods may be applied both in the presence and in the absence of a strong neutron background in the reactor.

In conclusion, the authors wish to express their sincere gratitude to F. B. Bryndin, A. S. Vodolazhskii, and V. S. Parshutin for taking part in the development and manufacture of the reactivity meter with the extensive dynamic range, and for participation in the measurements, and also V. N. Gurin and Yu. V. Volkov for valuable comments and advice when discussing the results of the measurements.

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CONTINUOUS ANALYSIS OF IODINE, CESIUM, BARIUM, STRONTIUM, YTTRIUM, AND RARE-EARTH ISOTOPES IN THE AQUEOUS COOLANT OF A NUCLEAR REACTOR

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The successful two-dimensional chromatographic separation of iodine and cesium isotopes from aqueous solutions [1] has made it possible to contemplate the continuous radiochemical analysis of the isotope composition of fission products in aqueous reactor coolants within the general monitoring of operation [2].

The investigations were made on the main loop of the MR reactor of the I. V. Kurchatov Institute of Atomic Energy. Figure 1 shows the scheme of the circuit arrangement. The basic features of the multisorbent block of continuous chromatographic separation are listed in Table 1 (the design and the principle of operation were described in [1]). A mixture of formic acid (to stabilize the ionic forms of the elements) and sodium chloride (to prevent the sorption of barium and strontium on ammonium molybdate) was used as the adjusting solution. The sections for preparing the sample and for radiochemical analysis and the draining collector were joined via flexible tubing with the line for taking samples, the special ventilating system, and the sewerage system. The flexible tubing between the equipment and the point at which samples were taken had a length of 15 m. The measuring chambers (100 ml for the I and Cs fraction and 200 ml for rare-earth elements — Y, Sr, and Ba) were joined with tubes of 2.5-mm diameter and 6-m length with the outlet openings of the intermediate collectors of the radiochemical analysis block and the draining collector of the sample-preparation block.

The γ radiation of the fractions was analyzed with a spectrometer equipped with a DGDK-25B semiconductor detector. The resolution of the spectrometer was 6.5 keV for 1332-keV γ radiation. The energy calibration had an accuracy of ± 4 keV in the range 0-1500 keV. The detector was placed 6 m from the separating block and no additional shielding was employed. The γ background was insignificant and originated mainly from the radiation of the isotopes ^{60}Co, ^{137}Cs, ^{18}F, and ^{24}Na. The background generated by ^{24}Na increased 1.5 times during the operation of the chromatograph.

During the tests, the setup was run through four cycles of continuous 10-h operation. Constant conditions of operation were maintained: the rotation period of the sorbent layers was 30 min; the flow rate of the initial solution and the eluting solution through the block of separation was 0.3-0.4 and 0.2-0.3 liter/h,

TABLE 1. Basic Features of the Radiochemical Analysis Block

Sorbent	Element separated	Eluting solution
Trioctylamin (TOA)	I	1 M NH_4NO_3
Di-2-ethylhexyl-orthophosphoric acid (D2ÉGFK)	REE and Y	0.01 M trilon B pH = 5.5
Ammonium phosphormolybdate (PMo)	Cs	5 M NH_4NO_3
Cationate (KU-2)	Ba and Sr	0.01 M trilon B, pH = 10

*REE = Rare-earth elements.

Fig. 1. Scheme of the setup for the con-
tinuous monitoring of the activity of I, Cs,
Ba, Sr, Y, and rare-earth elements: 1)
line for taking samples; 2) ceiling (between
floors); 3) special ventilation; 4) sample
preparation block; 5) block of continuous
radiochemical separation; 6) chamber for
the iodine fraction; 7) chamber for the
rare-earth element and Y fractions; 8)
chamber for the Cs fraction; 9) chamber
for the Ba and Sr fractions; 10) semicon-
ductor detector; 11) preamplifier; 12) low-
noise amplifier; 13) pulse analyzer; 14) digi-
tal printer; 15) special sewerage system.

respectively. In order to reduce the time of supplying a sample, a flow rate of 4-6 liters/h was maintained
in the loop consisting of the line for the removal of samples, the input collector of the sample-preparation
block, the draining collector, and the special sewerage system. The adjusting solution was added to the sam-
ple in the ratio 1 :4 in order to maintain in the sample 0.1 M concentrations of formic acid and ammonium
chloride. For the purpose of reducing the time within which the fractions separated were supplied to the detec-
tor, the rate at which the eluant was run through the sorbents with TOA, PMo, and KU-2* was doubled in the
course of five-day tests. The flow rate of the eluant through the layer with D2ÉGFK* was kept on the previous
level in view of the low activity of the isotopes to be monitored.

In the course of the entire testing period the reactor worked with constant power; in the loop uncorrected
neutral conditions with pH =7-7.5 were maintained; and the total specific activity of the dry residue was (3.5 -
4.5) $\cdot 10^{-4}$ Ci/liter 2 h after a sample had been taken. The specific isotope activity was determined with fast
chromatographic analysis [3] and is listed in Table 2 (the notation $m \cdot 10^{-n} = m - n$ was used).

During the continuous separation, seven series of γ radiation measurements were made on the fractions
"in flow"; two series of such measurements were made after the fractions had been introduced in the respec-
tive chambers. By analyzing the resulting spectra, $^{131-135}$I and ^{99m}Tc could be identified in the first fraction
(see Table 1), ^{91m}Y and ^{92}Y in the second fraction, ^{138}Cs in the third fraction, and $^{139-141}$Ba and $^{91-92}$Sr in the
fourth fraction. The values of the specific activity of the long-lived ^{131}I, ^{133}I, ^{135}I, ^{91}Sr, and ^{140}Ba (see Table 2)
isotopes and the data obtained for the corresponding isotopes from measurements made on the continuously
separated fractions were used to calibrate the detector for the photoefficiency in the case of a source geometry
in the form of 100- and 200-ml chambers. Based on these results, the specific activity of all isotopes moni-
tored was calculated for the moment of the measurement. The ratio of the activities calculated for the short-
lived ^{132}I, ^{134}I, ^{139}Ba, and ^{141}Ba isotopes for the moment of the measurement with the corresponding data of
Table 2 made it possible to determine the time within which the sample was transferred from the point of sam-
ple removal to the detector. This time amounted to about 60-70 min. When the flow rate of the eluting

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*The abbreviations are explained in Table 1.
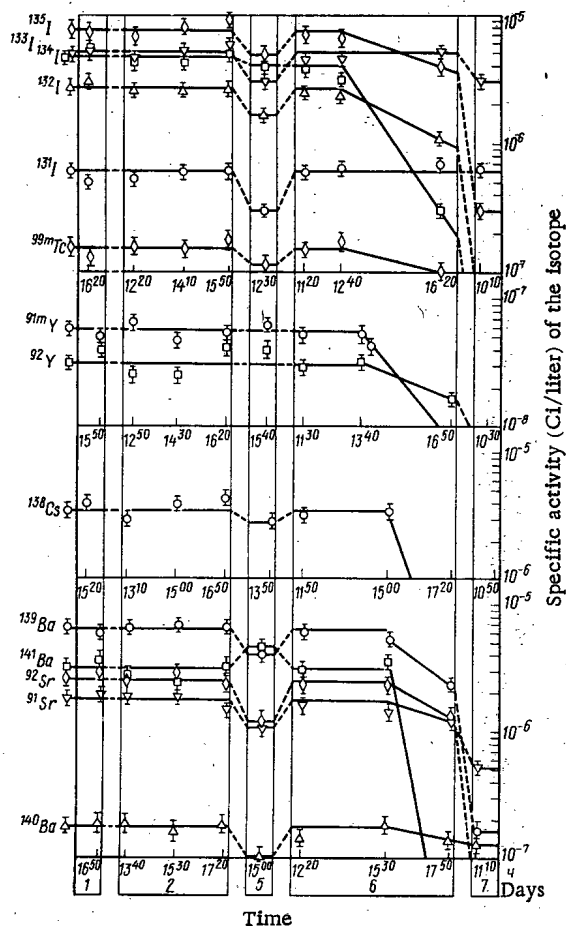
Fig. 2. Activity of the samples monitored in the measuring
chambers at various conditions of operation of the chromato-
graph.

solutions was doubled, the delay was reduced to 35-40 min. The dependence of the isotope activity upon both
the time and the rate at which the solutions flow through the measuring chambers is shown in Fig. 2.

A stable isotope composition in the fractions separated was observed during stationary operation of the
setup. The deviations of the experimental points from the average value did not exceed 15-20%; the ^{92}Y data
were an exception, which is explained by the low specific activity of ^{92}Y and the inherent insufficient statistical
accuracy of the measurements. The activity values of the yttrium isotopes, which are too low when compared
with the $^{91-92}$Sr mother isotopes, and the missing photopeaks which should originate from the radiation emitted
by radioactive cerium isotopes are apparently caused by the increased sorption of these elements on the

TABLE 2. Specific Activity (calculated for
the time at which the sample was taken) of
the Nuclides in the Coolant of the MR Re-
actor at the Beginning and at the End of
the Tests

Radio-	γ, Ci/liter		Radio-	ν.Ci/liter	
nuclide	first days	7th days	nuclide	first days	7th days
^{131}I	4,3—7	4,1—7	^{140}Ba	—	1,9—7
^{132}I	2,1—6	2,5—6	^{141}Ba	1,8—5	2,9—5
^{133}I	3,0—6	3,3—6	^{139}Ba	7,4—6	8,0—6
^{134}I	6,7—6	7,8—6	^{141}Ce	3,2—8	3,8—8
^{135}I	3,9—6	4,1—6	^{143}Ce	1,3—7	1,4—7
^{99m}Tc	2,0—6	1,5—6	^{144}Ce	6,0—8	7,0—8
^{138}Cs	1,3—5	1,1—5	^{91m}Y	1,0—6	1,3—6
^{91}Sr	2,5—6	2,0—6	^{92}Y	1,2—6	1,6—6
^{92}Sr	4,7—6	3,5—6	^{24}Na	2,0—4	2,2—4

surface of the construction materials. After trapping the fractions separated in the measuring chambers, the ratio of the activities of the isotopes monitored varied within the experimental accuracy limits in accordance with the half-lives of the elements. Specific conditions of a continuous separation of fractions could be established after repeated interruptions ranging from several dozen hours to several days. The flow rates of the eluting solutions could be used to adjust the activity ratio of the short-lived and long-lived isotopes in the fractions separated by the time of the measurement. For example, the activity ratio of ^{141}Ba and ^{92}Sr increased from 1.2 to 3.7 when the flow rate of the eluant was doubled.

A noticeable accumulation of long-lived radioactive isotopes in the measuring chambers and near the detector was not observed in the period of the tests. The background was almost constant in the course of all the tests. Products resulting from the corrosion of Cr, Mn, and Co, and also Zr and Ce were retained on a mechanical filter (porous teflon tablet). The absolute activity of the individual isotopes separated on the filter was approximately 10^{-8}-10^{-7} Ci.

Our results indicate that a continuous separation of I, Cs, Ba, Sr, and Y isotopes from the aqueous coolant of a nuclear power plant is feasible. Two conditions of operation of automated systems for monitoring the hermetic sealing of the fuel-element shells are possible: periodic, remote-control analyses of the radioactivity of isotopes in the coolant under conditions of stationary reactor operation, or continuous monitoring to assess the state of the fuel-element shells from fluctuations of the fission-product activity which arise when the conditions of reactor operation are modified.

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ACCELERATION OF IONS BY THE QUASI-STATIC
FIELD OF AN ELECTRON BEAM

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UDC 621.3.038.625

The following scheme of accelerating ions by the quasi-static field of an electron beam was proposed in [1, 2]. By sending a high-current electron beam through a corrugated metal tube situated in an external homogeneous magnetic field, one can generate a stationary longitudinal electric field which periodically depends upon the z coordinate in the direction of beam propagation. The maximum amplitude of the longitudinal field is given by the condition that the electrons stop and can be estimated with the formula $E_{zmax} \sim (\gamma_0 - 1)/L$, where γ_0 denotes the relativistic factor of the electrons; L denotes the modulation period of the system. With $\gamma_0 = 6$ and $L = 10$ cm, the formula renders the value $E_{z\,max} \simeq 50$ MV/m which considerably exceeds the accelerating field strengths of 1.5-2 MV/m obtained with conventional ion accelerators.

It was suggested to generate the travelling wave required for the acceleration of the particles with the aid of a slow, time-dependent modulation of the beam current at the input to the system (for $z = z_0$). By combining a spatial modulation (with the wave number $k = 2\pi/L$) with a time-dependent modulation (with the frequency $k_0 c$), one can generate a wave propagating with the phase velocity $v_{ph} = (k_0/k) c$. The parameters k and k_0 of the modulation are chosen so that, on the one hand, the required phase velocity $v_{ph} \simeq v_i$ (v_i denotes the velocity of the ions) is obtained, and, on the other hand, the rotational fields, which are proportional to the frequency and maintain a field pattern close to the stationary case, render a negligible contribution. The first condition means that the relation $k_0 = v_i k/c$ must be maintained; the second condition imposes restrictions upon $\beta_{ph}^2 = v_{ph}^2/c^2 = k_0^2/k^2 \ll 1$. Since in this scheme the accelerating field is generated with a high-current electron beam, the scheme might be successfully used to accelerate protons to medium energies at high proton intensities; the intensity is limited only insofar as the proton current must be much smaller than the electron current. In addition to that, the combination of large field gradients with the possibility of extensively adjusting the phase velocity of the accelerating wave (by changing k or k_0, very small β_{ph} values can be obtained) makes the scheme promising for the acceleration of heavy ions.

The goal of the present work is to determine the approximate requirements which the electron beam and the parameters of the system must satisfy and, more specifically, to establish a relation between the electron current, the degree of modulation of the boundary, and the longitudinal field.

We consider the stationary ($\partial/\partial t = 0$) state of an axially symmetric ($\partial/\partial\theta = 0$) monochromatic electron beam inside an ideal conducting tube which on the r, z plane of a cylindrical coordinate system has a periodic boundary in longitudinal direction, the boundary being described by the equation $F(r, z) = 0$. The beam is assumed to be infinite in z direction; the transverse motion of the beam is "frozen" by an external homogeneous magnetic field so that the electron trajectories are straight lines.

In hydrodynamics, the stationary state of such a beam with the temperature zero is given by the following equation system

$$\frac{1}{r}\frac{\partial}{\partial r}\left(r\frac{\partial\Phi}{\partial r}\right)+\frac{\partial^2\Phi}{\partial z^2}=4\pi\rho; \tag{1}$$

$$\frac{\partial}{\partial z}j_z=0; \tag{2}$$

$$\gamma_0=\gamma-e\Phi/m_0c^2, \tag{3}$$

where $\Phi(r, z)$ denotes the two-dimensional potential; $\rho(r, z)$ and $j_z(r, z)$ denote the charge density and the current density, respectively; $\gamma = (1 - \beta^2)^{-1/2}$, and $\beta = v/c$. Equation (1) is the Poisson equation, Eq. (2) is the continuity equation, and Eq. (3) is the law of energy conservation, where γ_0 has the meaning of the coordinate-independent total energy of a particle.
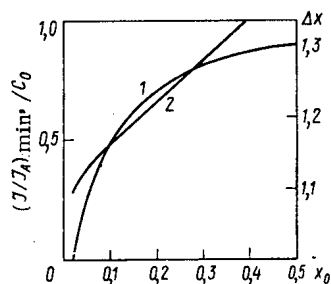
Fig. 1   Fig. 2

Fig. 1. Compact beam: 1) depth of the corrugation; 2) minimum current.

Fig. 2. Tubular beam for $\gamma_0 = 10$ and $x_0 = 0.1$; 1) charge per period of the system; 2) minimum current; 3) depth of the corrugation; 4) parameter of the charge-density modulation.

By eliminating from Eqs. (1)-(3) ρ and j_z, we obtain an equation which is the analog to the equation describing the formation of a virtual cathode (see, e.g., [3]):

$$\frac{1}{x}\frac{\partial}{\partial x}\left(x\frac{\partial \varphi}{\partial x}\right) + \frac{\partial^2 \varphi}{\partial \xi^2} = \frac{4\pi j_0}{k^2 J_A}\frac{\gamma_0 + \varphi(x, \xi)}{\sqrt{[\gamma_0 + \varphi(x, \xi)]^2 - 1}}, \qquad (4)$$

where $\varphi = e\Phi/m_0 c^2$, $x = kr$, $\xi = kz$, $J_A = m_0 c^3/e = 17$ kA, and $k = 2\pi L$.

Beyond the region occupied by the beam, the potential obeys the Laplace equation whose solution must satisfy the boundary conditions on the surface of the conducting tube

$$\varphi(r, z) = \text{const} \quad \text{for} \quad F[r(z), z] = 0, \qquad (5)$$

and must also satisfy the conditions of matching with the solution of Eq. (4) on the boundary between the beam and vacuum (if the beam is tubular, then matching on two boundaries must take place).

Serious mathematical difficulties are encountered in the solution of the straightforward problem, when the self-consistent field in the beam is to be determined along a given boundary. However, in the case under consideration one can treat the inverse problem, i.e., one can start from some potential distribution which is periodic in z direction and satisfies the corresponding differential equations and matching conditions and then use Eq. (5) to determine the shape of the boundary surface which ensures the specific potential distribution. We use the inverse approach and consider the stationary state of compact and tubular beams.

Compact Beam. Assume a cylindrical beam of radius r_0 which satisfies the conditions formulated above and which is composed of relativistic particles for which

$$\gamma^2(x, \xi) \gg 1. \qquad (6)$$

Condition (6) makes it unnecessary to consider the self-consistent field and allows us to disregard the influence of the intrinsic fields upon the velocity of the particles in the beam. The problem is reduced to determining the field generated by a charged rod which is homogeneous in r and z direction and enclosed by a periodic boundary. In this case a longitudinal field results from the curvature of the boundary, because the modulation of the charge density in z direction is negligibly small in the beam.

Equation (4) assumes the following form

$$\frac{1}{x}\frac{\partial}{\partial x}\left(x\frac{\partial \varphi}{\partial x}\right) + \frac{\partial^2 \varphi}{\partial \xi^2} = \frac{4J}{J_A x_0^2}; \quad J = \pi r_0^2 j_0; \; x_0 = kr_c. \qquad (7)$$

The solution to this equation, which is periodic in ξ and represents a finite potential on the axis, is written in the form

$$\varphi(x, \xi) = C_0 I_0(x)\sin\xi - \frac{J}{J_A}\left[1 - \left(\frac{x}{x_0}\right)^2\right] - C_0 I_0(x_0), \quad x \leqslant x_0, \qquad (8)$$

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where $I_0(x)$ denotes a Bessel function of imaginary argument; C_0 denotes the dimensionless amplitude of longitudinal modulation of the potential; J/J_A and C_0 must satisfy the condition

$$J/J_A + 2C_0 I_0(x_0) \ll \gamma_0 - 1 \tag{9}$$

in accordance with condition (6).

The longitudinally periodic solution to the Laplace equation for the potential beyong the beam assumes the form

$$\varphi(x, \xi) = \frac{2J}{J_A} \ln \frac{x}{x_0} + C_0 I_0(x) \sin \xi - C_0 I_0(x_0), \tag{10}$$

with the matching conditions (8) for $x = x_0 = kr_0$. We omitted the term describing the field originating from the variable charge density component which can be ignored in the case under consideration.

Equations for the equipotentials can be easily derived from Eqs. (8) and (10). An investigation of the structure of the equipotentials for a given x_0 shows that for small $\alpha = (J/J_A)/C_0 I_0(x_0)$ the equipotential lines are closed only for $r \to \infty$. Closing on a finite radius begins when the parameter α reaches some value α_1 but as long as $\alpha_1 \leq \alpha < \alpha_2$, only equipotential lines which are fully or partially inside the beam close; a metal boundary cannot be arranged along these equipotential lines. Finally, for $\alpha = \alpha_2$ the first equipotential line which is situated fully outside the beam closes. The equation of this equipotential is written as

$$2\alpha \ln \frac{x}{x_0} + \frac{I_0(x)}{I_0(x_0)} \sin \xi - 1 = 0. \tag{11}$$

A further increase in α causes the closing of an increased number of equipotential lines situated outside the beam; in principle, a conducting surface could be arranged along these equipotential lines. But a given field strength E_{0z} is reached with a minimal beam current if the boundary follows the equipotential described by Eq. (11). Then

$$(J/J_A)_{\min} = \alpha_2 C_0 I_0(x_0). \tag{12}$$

Apart from this, this boundary is more advantageous as far as the energy is concerned, because in this case the constant component of the potential difference between the edge of the beam and the wall is minimal.

The minimum current required for generating a given longitudinal field strength E_{0z} can be calculated with the dependence of $(J/J_A)_{\min}/C_0$ and of the degree of modulation $\Delta x = x_{\max} - x_0$ upon the dimensionless beam radius x_0 (dependence shown in Fig. 1). The geometrical dimensions of the corrugation are automatically chosen so that a sinusoidal potential distribution in the z direction is obtained. In particular, for a beam with radius $r_0 = 0.5$ cm, a field strength of 10 MV/m requires a current $J/J_A = 0.48$ ($C_0 = 1$). The period of modulation L of the boundary is in this case 31.4 cm, and the depth of the modulation is $\Delta r = 5.8$ cm. A beam of the same radius but with a current $J/J_A = 1$ renders a field strength $E_{0z} = 40$ MV/m at $L = 7.85$ cm and $\Delta r = 1.6$ cm. We obtain in the cases under consideration from condition (9) $\gamma_0 \gg 4$ for the total energy.

Thus, an electron beam of uniform cross section does not render the above-estimated maximum fields and, in addition, requires a rather high power ($J \sim J_A$, $\gamma_0 \gg 4$). This is basically related to the large transverse "creeping" of the potential in the beam, because the transverse electric field is much stronger than the longitudinal electric field at reasonable values of the modulation. As far as the acceleration aspect of this scheme is concerned, tubular beams offer greater possibilities [4].

Tubular Beam. We consider within the above model an infinitely thin tubular beam of radius r_0. In this case the external potential φ_{II} (for $x \geq x_0$) and the internal potential φ_I (for $x \leq x_0$) satisfy the Laplace equation and the conditions of matching on the beam surface (for $x = x_0$):

$$\varphi_I(x = x_0, \xi) = \varphi_{II}(x = x_0, \xi) = \varphi(\xi); \tag{13}$$

$$\frac{\partial \varphi_{II}}{\partial x}(x = x_0, \xi) - \frac{\partial \varphi_I}{\partial x}(x = x_0, \xi) = 4\pi\sigma(\xi), \tag{14}$$

where $\sigma(\xi)$ denotes the surface charge density for which Eqs. (2) and (3) render the formula

$$\sigma(\xi) = \frac{J_k}{2\pi x_0 c} \frac{\gamma_0 + \varphi(\xi)}{\sqrt{[\gamma_0 + \varphi(\xi)]^2 - 1}}. \tag{15}$$

The potential distribution $\varphi(\xi)$ on a surface covered by the beam can be obtained with φ_I or φ_{II}. When we normalize the potential so that $\varphi_{I, II}(x = x_0, \xi = \pi/2) = 0$, we have for $\varphi(\xi)$

$$\varphi\,(\xi) = b\sin\xi - b, \tag{16}$$

where b denotes some constant having a value of the interval ranging from 0 to $(\gamma_0 - 1)/2$.

Tremendous calculations would result from substituting Eq. (15) into Eq. (14). For estimates we can use an approximation formula which renders the same degree of modulation of the charge density as Eq. (15):

$$\sigma\,(\xi) = \frac{J_h}{2\pi x_0 c}\,(Q - R\sin\xi), \tag{17}$$

in which

$$Q = \frac{1}{2}\left[\frac{\gamma_0 - 2b}{\sqrt{(\gamma_0 - 2b)^2 - 1}} + \frac{\gamma_0}{\sqrt{\gamma_0^2 - 1}}\right]; \tag{18}$$

$$R = \frac{1}{2}\left[\frac{\gamma_0 - 2b}{\sqrt{(\gamma_0 - 2b)^2 - 1}} - \frac{\gamma_0}{\sqrt{\gamma_0 - 1}}\right]. \tag{19}$$

With Eqs. (13), (14), and (17), we obtain the following formula for $\varphi_{\mathrm{II}}(x,\,\xi)$:

$$\varphi_{\mathrm{II}}\,(x,\,\xi) = \frac{2J}{J_A}Q\ln\frac{x}{x_0} + \left\{\left[\frac{b}{I_0(x_0)} - \frac{2J}{J_A}RK_0(x)\right]I_0(x) + (2J/J_A)\,RI_0(x_0)\,K_0(x)\right\}\sin\xi - b, \tag{20}$$

where $K_0(x)$ denotes the Macdonald function.

The structure of the equipotentials described by Eq. (20) is analogous to that of the compact beam and makes it possible to derive the relations between the parameters of the system:

$$I_1(x_{\max})\ln\frac{x}{x_{\max}} - [I_0(x_0) + I_0(x_{\max})]/x_{\max}$$
$$= \frac{R}{Q}I_0(x_0)\left[\frac{1}{x_{\max}} + K_0(x_0)\,I_1(x_{\max})\right.$$
$$\left. + I_0(x_0)\,K_1(x_{\max})\right]; \tag{21}$$

$$(J/J_A)_{\min} =$$
$$= \frac{1}{2}\frac{bI_1(x_{\max})/I_0(x_0)}{\dfrac{Q}{x_{\max}} + R\,[K_0(x_0)\,I_1(x_{\max}) + I_0(x_0)\,K_1(x_{\max})]}; \tag{22}$$

$$\mu = \sigma_{\max}/\sigma_{\min} = (Q + R)/(Q - R); \tag{23}$$

$$\frac{r_e}{L}\nu_L = Q\,(J/J_A)_{\min}: \tag{24}$$

$$E_{0z}L = 2\pi b m_0 c^2/eI_0(x_0), \tag{25}$$

where $x_{\max}$, E_{0z}, and $(J/J_A)_{\min}$ denote the same quantities as in the case of the compact beam; ν_L is proportional to the number of particles falling into one period L of the system; $r_e = e^2/m_0c^2$ denotes the classical electron radius; and μ denotes a coefficient characterizing the degree of modulation of the charge density.

The results of calculations made with these formulas are shown in Fig. 2. The ascending branches of the curves referring to the minimum current and the total charge per period indicate the state of a beam with low density modulation $(\mu \approx 1)$; increasing fields strengths are obtained by increasing the charge (the current). The descending sections of the curves correspond to a state with strong modulation $(\mu$ is large); the field strength increases even when the total charge (current) decreases. At a particular beam current, two stationary states of the beam can develop with different degrees of charge-density modulation in the beam. This accounts for the ambiguity of the field-strength dependence upon ν_L (the current).

The modulation coefficient μ depends upon the depth Δx of the corrugation. For $\Delta x \to \Delta x_{\mathrm{crit}}$ the field strength approaches the maximum which in this model is $E_{0Z}L = \pi m_0c^2(\gamma_0 - 1)/eI_0(x_0)$. But then the beam resembles a virtual cathode, so that the problem of the stability of such a state requires special investigations [5].

Let us state some numerical values of the system parameters. Assume, as in the case of the compact beam, $x_0 = 0.1$ and $r_0 = 0.5$ cm. Then for a current $J/J_A = 0.48$ in the stationary state of a tubular beam with weak modulation we obtain a field strength $E_{0z} \simeq 10$ MV/m (this value coincides with the value of the field strength generated by the compact beam with the same parameters). A stronger field, $E_{0z} \simeq 43$ MV/m, is obtained in the stationary state with the same current but with strong modulation, as could be anticipated. The

depth of the corrugation is 5.8 and 9 cm for beams with weak and strong charge-density modulation, respectively; the respective modulation coefficients of the charge density are 1 and 3.

These estimates show that an experimental investigation of the method can be initiated at the present time, because the technique of generating kiloampere electron beams with particle energies of several megawatts has been mastered in the nanosecond range [6] and is being successfully developed in the microsecond range.

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COMPENSATION OF A SPACE-LIMITED POSITIVE CHARGE BY ELECTRONS

V. M. Kulygin and V. I. Telegin   UDC 533.9

It is well known that a necessary condition for transmitting an intense ion beam is the compensation of its space charge. A positive-ion charge is usually compensated by means of electrons, which are either produced as the result of ionization of the residual gas by particles of the beam itself, or are specially introduced from outside. In practically every case a space-limited system of ions and electrons is obtained.

Calculations of the compensation of the positive charge of an ion beam have, up till now, been carried out either in the geometrical approximation [1], or by solving the equations of motion of the individual particles in a

Fig. 1. Oscillations of electron density at the coordinate origin for V_T equal to: 1) 0.020; 2) 0.025; 3) 0.030; 4) 0.035.

Fig. 2. Charge distribution at differ-
ent times τ.

self-consistent field [2]. In the first case it has been possible to calculate some average parameters of the system, by giving a specific form to the distribution functions. In the second case the interpretation of the results often depends on the method of averaging the self-consistent field, for example.

Numerical solutions of Vlasov's equation use the kinetic approach for treating the model problem of space-charge compensation.

The present paper considers the motion of the compensating electrons, and in particular, the effect of inhomogeneity on this motion. Assuming that the system parameters change much more slowly along the beam axis than across it, we restrict ourselves to studying transverse motion only. Thus, excitation of electron oscillations by beam ions is outside the scope of the present treatment. Finally, remembering that $m_i \gg m_e$, where m_i and m_e are the ion and electron masses, respectively, we neglect transverse motion of the ions.

Fundamental Equations. We have a one-dimensional bounded plasma in the region $0 \le x \le L$, consisting of immobile ions and electrons which compensate the space charge of the ions. The density of ions is a given function of the coordinates $N = N(x)$

$$N(x) = \begin{cases} \left[1 - \left(\frac{x}{R} \right)^2 \right]^2, & 0 \leqslant x \leqslant R; \\ 0 & R \leqslant x \leqslant L; \end{cases} \tag{1}$$
$$R = (L; \ 0.8L).$$

The behavior of the electrons is described by Vlasov's equation

$$\frac{\partial f}{\partial t} + v \frac{\partial f}{\partial x} - \frac{eE}{m} \frac{\partial f}{\partial v} = 0, \tag{2}$$

where f is the electron distribution function, v is the velocity of an electron, e is the electronic charge, and E is the electric field which has the following boundary conditions:

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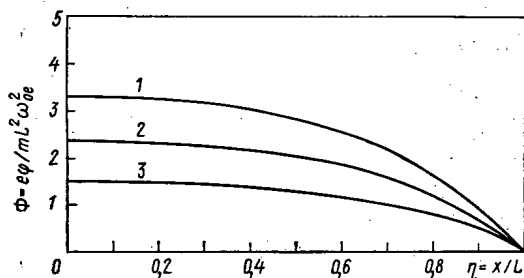
Fig. 3   Fig. 4

Fig. 3. Spatial distribution of the average potential for VT equal to: 1) 0.030; 2) 0.025; 3) 0.020.
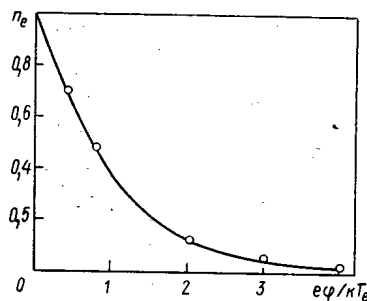
Fig. 4. Boltzmann electron distribution. The curve is drawn from Boltzmann's formula, and the points give the results of calculations.

If $x = 0$ (in the symmetry plane of the beam), the electrons are reflected

$$f(0, v, t) = f(0, -v, t); \tag{3}$$

if $x = L$ (close to the wall), electrons with positive velocity are absorbed, i.e.,

$$f(L, v, t) = 0 \quad \text{for} \quad v < 0. \tag{4}$$

The electric field is determined by Poisson's equation

$$\frac{\partial E}{\partial x} = 4\pi e \left\{ N(x) - \int_{-\infty}^{+\infty} f(x, v, t)\, dv \right\} \tag{5}$$

and the boundary condition

$$E(0, t) = 0. \tag{6}$$

The initial electron distribution function is specified to have the form

$$f(x, v, 0) = \frac{n_0(x)}{\sqrt{\pi}\, v_{T_e}} \exp\left[-\frac{v^2}{2 v_{T_e}^2}\right], \tag{7}$$

where v_{T_e} is the thermal velocity of electrons and n_0 is the electron density at the initial moment of time

$$n_0(x) = \Gamma N(x), \tag{8}$$

where Γ is a parameter characterizing the degree of compensation of ion space charge by electrons. In the present paper the basic calculations were carried out for $\Gamma = 1$. The following dimensionless variables were used in the calculations:

$$\tau = t\omega_{0e}; \quad \eta = \frac{x}{L}; \quad V = \frac{v}{L\omega_{0e}}; \quad \Phi = \frac{e\varphi}{mL^2\omega_{0e}^2};$$

$$F = \frac{e}{m}\frac{E}{L\omega_{0e}^2}.$$

Here $\omega_{0e} = (4\pi e^2 n_0/m)^{1/2}$ is the electron plasma frequency, connected with the electron density at the coordinate origin for $t = 0$, $E = -\partial\varphi/\partial x$. In the variables adopted, Eq. (1) takes the form

$$\frac{\partial\psi}{\partial\tau} + V\frac{\partial\psi}{\partial\eta} - F\frac{\partial\psi}{\partial V} = 0, \tag{9}$$

$$\psi = \psi(\eta, V, \tau); \quad \int_{-\infty}^{+\infty} \psi\, dV = \frac{1}{n_0}\int_{-\infty}^{+\infty} f\, dv,$$

while Poisson's equation is rewritten as

$$\frac{\partial E}{\partial\eta} = \xi(\eta) - \int_{-\infty}^{+\infty} \psi\, dV, \tag{10}$$

where $\xi(\eta) = N/n_0(0)$. The corresponding boundary conditions are written as

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$$\psi(0,\ V,\ \tau) = \psi(0,\ -V,\ \tau);$$
$$\psi(1,\ V,\ \tau) = 0,\ (V < 0);$$
$$F(0,\ \tau) = 0.$$

(11)

Results of the Calculations. The system of equations given above, together with the boundary and initial conditions, was solved for various values of the parameters $V_T \equiv v_{Te}/L\omega_{0e} \sim d/L$ and Γ. Here $d = (T_e/4\pi e^2 n_0(0))^{1/2}$ is the Debye radius at the coordinate origin, T_e is the electron temperature, V_T varied in the range from 0.02 to 0.1, and Γ from 0.987 to 1.2.

The time variation of electron density at the coordinate origin is given in dimensionless quantities for various thermal velocities with $\Gamma = 1$. The spatial distribution of charge density $q(x) = e[N(x) - n(x,\ t)]$ given in Fig. 2 for different times τ, illustrates the structure of the resulting oscillations. The resulting spatial distribution of potential can be seen in Fig. 3. Many variants of the calculations give a maximum value of the potential at the coordinate origin of $(3.7 - 3.8)\,T_e/e$. Fluctuations of potential and electron density at the coordinate origin do not exceed 1-2%.

If $\Gamma \geq 1$ at the initial moment, i.e., total compensation or excess of electrons in the beam, then all the "surplus" electrons are ejected to the wall in a time L/v_{T_e}. If $\Gamma < 1$, but $e\varphi > 3.8T_e$, there is practically no ejection of electrons, and a Boltzmann distribution is established in the system (Fig. 4).

Discussion of Results. The form taken for the initial distribution function differs from the equilibrium distribution established in the calculations. In this connection electron density oscillations arise around the equilibrium state. At first they are of large amplitude, subsequently they decrease considerably, and later on remain at a certain level, which is larger the higher the electron temperature.

The electron density oscillations under consideration clearly exhibit rapid oscillations of period 2π at the point $\eta = 0$, which corresponds to the period of plasma oscillations in the time scale adopted.

Since $k \sim 1/L$, the quantity kd (where k is the wave number) turns out to be proportional to V_T. Thus, in the present problem the inequality $k^2 d^2 \ll 1$ always, which in linear theory is the condition for weakly damped plasma waves [3]. Thus, the sharp decrease in their amplitude in the process of reaching steady state must be connected with nonlinear effects.

Apart from plasma waves, slower perturbations of electron density can be seen. Their frequency depends on the parameters V_T and Γ and varies from (1/12) to (1/50)ω_{0e}. It is not difficult to conclude from the phase shift of these oscillations at neighboring points in space that their phase velocity is proportional to the electron thermal velocity. By their nature these oscillations are the oscillations of electrons in a potential well formed as a result of the thermal motion of electrons in a bounded plasma.

In fact, if we take the equilibrium electron density distribution to be Boltzmann

$$\bar{n}_e = \bar{n}_0 \exp\left[-\frac{e\varphi}{T_e}\right],$$

(12)

then the potential well has the depth

$$\Delta\varphi = \frac{T_e}{e}\ln\frac{\bar{n}_0}{\bar{n}_{e\,b}},$$

(13)

where $\bar{n}_0$ and $\bar{n}_{e\,b}$ are the time averaged equilibrium electron densities at the center and at the wall, respectively.

The oscillation frequency of electrons in the potential well can be estimated as

$$\omega^2 \sim \frac{\Delta\varphi e}{a^2 m},$$

(14)

where a is the characteristic dimension of the well. Taking Eq. (13) into account we have for $a \sim L$

$$\omega \sim \frac{v_{Te}}{L}\left[\ln\frac{\bar{n}_0}{\bar{n}_{e\,b}}\right]^{1/2}.$$

(15)

Thus, increasing the depth of the potential well, both by increasing v_{T_e}, as well as by introducing incomplete compensation of the ion space charge, should lead to an increase in the frequency of the oscillations under consideration, which is in qualitative agreement with the results of numerical analysis. Remembering that $k \sim 1/L$, we obtain the phase velocity of wave propagation

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$$v_{\mathrm{p}} \sim v_{T_e} \left[\ln \frac{\bar{n}_0}{\bar{n}_e} \right]^{1/2}. \tag{16}$$

The proportionality of phase velocity to the electron thermal velocity also results from the numerical analysis.

Brief Conclusions. Numerical investigations carried out on the model adopted above showed that in the motion of electrons compensating space-limited ion volume charges there exist oscillations at frequencies close to the electron plasma frequency, and also close to the frequency $\omega \sim v_{T_e}/L$. These last oscillations can be identified with the oscillations of electrons in a potential well formed by the ion charge. The depth of this potential well is $\sim (3.7\text{--}3.8)T_e/e$.

The steady-state mean electron distribution function in the system under investigation is close to a Boltzmann distribution.

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NUCLEAR REACTIONS IN THE SUN

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Solar physics has experienced a profound shock since the observation of R. Davis et al., disclosed in 1968, that the solar neutrino flux being recorded is less than that expected by an order of magnitude. While preparations for the neutrino observations were going on, calculations of solar models were being vigorously conducted. Similar results were obtained in different papers; therefore, the impression was formed that all was well in the theory of the sun. But the first observations gave an unexpected result — there were very few neutrinos. Attempts to reconcile the calculations with the observations gave rise to a series of papers in which the physical principles and astrophysical assumptions used in calculations of solar models were subjected to a new analysis. Of the physical data, the cross sections of the reactions which lead to the formation of the neutrino sources were first of all subjected to revision. Among these sources ^8B, which is formed in the reactions $^4He(^3He, \gamma)^7Be$, $^7Be(p, \gamma)^8B$, was assumed to be the most effective. Although the role of these reactions in the energy supply of the sun is slight, the large energy of the ^8B neutrinos (up to 14 MeV) makes them the most effective for the chlorine neutrino detector of the Davis setup, in which the decay of the ^{37}Ar formed in the reaction $^{37}Cl(\nu, e^-)^{37}Ar$ is recorded.

The formation of ^4He from four protons is necessarily accompanied by the emission of two neutrinos regardless of the method of synthesis. But the energy spectrum of the neutrinos depends on the method, and the possibility of recording neutrinos depends on the relative yield of ^8B, which the greater it is, the higher the concentration of ^3He and ^7Be and the higher the temperature. The yield of the reaction $^7Be(p, \gamma)$ depends strongly on temperature ($\sim T^{13}$).

The reaction $^7Be(p, \gamma)^8B$ is the least reliably investigated of the reactions which lead to the formation of ^8B. Measurements of the reaction cross section are rather difficult: ^7Be is radioactive (half-life of 53 days); therefore, it is complicated to produce in nuclear reactions and difficult to accumulate in a large amount. In addition, the decay of ^7Be is accompanied by γ radiation with an energy of 0.48 MeV, which hinders observation of the reaction $^7Be(p, \gamma)^8B$. Experimental data on the reaction cross section have been obtained only at rather low energy (~ 0.5 MeV). Moreover, the so-called gamma-peak of the reaction probability is at an energy ~ 3 keV. This means that a large extrapolation of the cross section in the very-low-energy region is necessary for the calculations. Such an extrapolation is all the more uncertain due to the fact that the variation of the cross section is complicated in the region which has been experimentally examined by a resonance at an energy of 0.75 MeV. Although different authors give results which differ by no more than a factor of two, this still does not indicate that the true result lies among them. The problem of the reliable determination of the cross section of the $^7Be(p, \gamma)^8B$ reaction at an energy of about 3 keV is considered practically unsolved. If the true cross section of the reaction is less than the assumed value, then the disagreement of the calculations with Davis' observations is smaller.

An increase in the probability of the concurrent reaction $^3He + ^3He \rightarrow ^4He + 2H$ may work in the same direction. This reaction terminates the main branch of the hydrogen cycle of nuclear synthesis. The ^3He concentration depends strongly on the cross section of this reaction. The larger the cross section is, the less ^3He there is and the less probable is the incidental reaction $^4He + ^3H \rightarrow ^3Be + \gamma$, which leads to the formation of ^8B. Therefore, the suggestion has been made of a resonance maximum in the interaction cross section of ^3He nuclei at low energy. A communication [1] has appeared concerning attempts to find this resonance by means of observing the spectrum of the neutrons of the reaction $^4He(^3He, n)^6Be$. The results are negative. But the searches are probably not ended, and the investigation of the $^3He - ^3He$ interaction at low energy remains an important problem in nuclear physics.

Revision of nuclear physics data with the goal of explaining the disagreement of the calculations with the observations of Davis has not given satisfactory results [2]. Although the new probable value for the flux has turned out to be at least two times smaller than that expected prior to the observations, the discrepancy has been preserved — the observed flux is less than the calculated probable value by a considerable factor. Therefore, a series of qualitatively extraordinary hypotheses which go beyond the framework of the known has appeared. For example, the suggestion [3] of quark catalysis of nuclear reactions has been put forward. Let us assume that relict quarks having negative charge have been preserved in the solar interior. Forming quasi-atoms and quasi-molecules with protons and in this way binding protons to each other, they catalytically accelerate the synthesis of deuterium from hydrogen and lower the temperature and energy of the neutrinos. This is required for agreement with Davis. A similar catalytic effect on nuclear reactions has already been discovered long ago in the case of the capture of negative muons by isotopes of hydrogen. However, it was found [4] that a fractionally charged quark cannot combine two protons into a molecule; a negative charge > 0.94e in absolute value is necessary; therefore, quarks with charge $(-1/3)e$ and $(-2/3)e$ do not help the synthesis even if they are in the sun.

Similar ideas have given rise to a hypothesis that a heavy neutral boson exists inside the sun [5]. If such a boson exists and interacts with neutrinos 10^8 times more strongly than owing to the usual known weak interaction, then by accelerating the synthesis of hydrogen it reduces the energy of the neutrinos and results in agreement with Davis' observations. Experiments in which attempts were made to find confirmation of these ideas have been published, and positive results have even been obtained [6]. Deuterons from the reaction $^3\text{He} + ^3\text{He} \rightarrow ^4\text{He} + ^2\text{D} + e^+ + \nu$ were sought. It has been confirmed that such deuterons are detected, and their appearance indicates the existence of an interaction which is significantly stronger than the known weak interaction. Evaluating these results, one can say that their conclusiveness is far from being sufficient to confirm such an unexpected idea. If more serious attempts were to confirm the hypothesis, this would be one of the outstanding discoveries of contemporary physics. But this is not very likely.

One of the ways people have sought to explain the disagreements with the observations is in diverse variations of the properties of solar neutrinos. If neutrinos change their properties on the way from the sun to the earth, for example, if they decay and become incapable of causing a reaction with chlorine in the detector, then the disagreement is removed. Several versions of neutrino decay have been suggested. Thus B. Pontecorvo [7] has suggested the conversion of an electron neutrino into a muon neutrino. J. Bahcall et al. [8] have proposed the conversion of a solar neutrino into a new neutrino and a photon. Even more complex hypotheses have been advanced about the mass spectrum of different neutrinos [9].

The number of similar hypotheses is probably unlimited and may be greater, the more unexpected are the ideas at their basis. Therefore, it is not surprising that black holes have not escaped the fate of proposed participants in internal solar processes. The appearance of such hypotheses indicates the fundamental character of the problem in solar physics on the one hand and its very inadequate soundness on the other.

The astrophysical trends of revision of the solar models are more realistic. All previous models were based on the steadiness of the processes in the solar interior. The external photon luminosity of the sun is rather stable. This is already reflected by the term "solar constant" assigned to the flux of solar energy at the earth $(1.388 \cdot 10^6 \text{ ergs} \cdot \text{sec}^{-1} \cdot \text{cm}^{-2})$. The well-known phenomena of surface solar activity are weakly reflected in its total luminosity, whose recorded fluctuations do not exceed 0.5-1.5%. It is not surprising that the first models of the sun assumed the steadiness of the processes throughout its entire interior. Convective energy transport and mixing of material was assumed only in a layer about 0.1 radius in thickness close to the surface. Energy transport by photon diffusion only was considered in the deeper layers.

The radiative energy transport time from the center to the surface is tens of millions of years. If some kind of unstable processes with variations in the energy release occur in the solar interior, then they are smoothed out on the way to the surface. Therefore, the stability of the surface luminosity of the sun is completely compatible with an instability in its interior if the time scale of this instability is small in comparison with tens of millions of years.

Some evidence of internal instability has been obtained in observations of solar flares [10]. The isotopic composition of the helium was surprising — too large an excess of ^3He. A certain amount of the excess ^3He could be formed at the surface in nuclear reactions, which occur here during flares. But the observed excess cannot possibly be explained in this way; it is too great. The conclusion that material is carried out from deep layers in which there is a lot of ^3He suggests itself. It has usually been assumed that the maximum ^3He content should lie at a depth of about one-half the radius. If material from this depth percolates to the surface, mixing touches upon significantly larger amounts of the material than does the convective zone assumed up till

now. Recent observations of neutrinos by Davis have revealed some increase in the number of recorded events. It is true that the accuracy of these observations is still not high enough, but if they are confirmed, one will be able to consider them as evidence of non-steadiness in the sources of the solar neutrinos.

If unforeseen mixing processes occur near the center of the sun, there where nuclear burning is going on, the composition of the "fuel" may differ significantly from what has been assumed. In particular, the ^{3}He concentration will prove to be significantly higher. And this means that the relative role of the non-neutrino reaction ^{3}He + ^{3}He → ^{4}He + 2p is increased, since its probability is proportional to the square of the ^{3}He concentration, while the yield of the reaction ^{3}He + ^{4}He → ^{7}Be + ν depends linearly on the ^{3}He concentration (the ^{4}He content changes more mildly). An increase in the ^{3}He concentration near the center and the temperature decrease associated with it lead to a decrease in the probability of the formation of the energetic ^{8}B neutrinos. If ^{8}B is not formed at all, only the synthesis of deuterium from hydrogen serves as a source of neutrinos. The observations of Davis do not contradict the assumption that all the solar neutrinos have a spectrum corresponding to the synthesis of deuterium. But it is difficult to exclude altogether the formation of ^{8}B in the existing models; therefore, ways are being sought to decrease its role.

The mixing of material in the center of the sun turns out to be rather effective even on small scales [11]. Differential rotation of the sun is discussed as the physical cause of the mixing. It is known from observations of the surface of the sun that the period of its rotation at the equator is ~26 days and near the poles is ~37 days. The rate of rotation evidently varies with the depth of a layer. There is an assumption that the inner layers are rotating more rapidly than the surface layers. Such a nonuniform (non-solid body) rotation is called differential rotation. It is assumed that the rotation gradually builds up to instability whose discharge results in mixing, which alters the composition of the material in the active zone. Therefore, the energy release at the center and the neutrino luminosity of the sun are not constant and vary in a characteristic time of several million years. If Davis' observations coincided with a minimum in the neutrino luminosity of the sun, then their results are not surprising. Terrestrial confirmations of a variable solar luminosity are being sought and found in the periodic ice ages on the surface of the earth.

All the searches for solutions of the problem of the evolution and energy supply of the sun are still far from completion and conclusiveness. But there are still no serious reasons to doubt the nuclear nature of the sources of solar energy. The neutrino flux expected from the synthesis of hydrogen into helium without the formation of ^{8}B does not contradict Davis' observations.

It is probable that the development of astrophysical premises in the construction of solar models opens up completely realizable approaches for approximating the truth. A new phenomenon, significant for solar physics, has quite recently been discovered. The observations of A. B. Severnyi at the Crimean Observatory have shown that the surface of the sun oscillates with a period of 160 minutes. This fact indicates that mechanical processes propagating with the speed of sound, i.e., sun-shaking waves, exist in the solar interior, perhaps right down to the very center. If these sound waves transport energy, the heat exchange between the center and the outer layers of the sun may differ significantly from what has been assumed, since the time for sound waves to pass from the center to the surface (of the order of half an hour) is less by many orders of magnitude than the tens of millions of years required for the transport of energy by radiation. The attention of observers and theoreticians is presently attracted to investigations of sun-shaking. Sun-shaking with various periods up to an hour has been detected by other observers. It is probable that solar physics will experience significant changes in the next few years.

Although the most realistic approaches to the resolution of the disagreements may prove to be by astrophysical means, nuclear physics is still in debt to solar physics. In the first place, conclusive and more accurate data about the cross sections of nuclear reactions are necessary, and secondly, it is desirable to refine Davis' data, which have thus far barely exceeded the background. In particular, research on new neutrino detectors is very important to develop ones which are distinguished by a lower threshold than chlorine and are able to record neutrinos of the main branch of thermonuclear synthesis: the p + p → D + e$^+$ + ν^- and p + p + e$^-$ → D + ν^- reactions. Evidently, the next detector after the chlorine one is the gallium detector, in which the reaction ^{71}Ga(ν, e$^-$)^{71}Ge with a threshold of 0.231 MeV is used. Research with it has been going on for a long time now and is being conducted at the P. N. Lebedev Institute of Physics of the USSR Academy of Sciences.

Although Davis' observations have turned out to be in disagreement with the usual solar models, there are no serious reasons yet to doubt that thermonuclear hydrogen burning with helium formation is the source of solar energy. The chlorine detector of Davis is not sensitive to neutrinos of nonboron synthesis; therefore, measurements with low-threshold detectors are very important. The total flux of solar neutrinos is connected rather exactly to its luminosity: a pair of neutrinos is emitted for each 25 MeV of solar energy. Their softest

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spectrum is characteristic of the nonboron hydrogen cycle of synthesis reactions. If there are fewer such neutrinos than expected, the necessity of a thorough revision of solar physics and perhaps of stellar physics arises.

The successes of stellar theory based on the analysis of the thermonuclear synthesis of the light elements are presently so varied that the correctness of their bases is not a cause for doubt. Observations of solar neutrinos are very necessary for establishing the truth of the hypothesis that thermonuclear reactions are the sources of the energy radiated by the stars.

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RADIATION SAFETY AT NUCLEAR POWER STATIONS*

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The main factor in radiation effects on personnel at all operating nuclear power stations is external γ radiation; the contribution from it ordinarily makes up more than 95% of the external radiation dose. According to the data in [1], the mean annual radiation dose for controlled personnel at the Novovoronezh nuclear power station was 0.6-0.9 rem/yr during the period 1972-1974. There were no cases of doses greater than the established limit (5 rem/yr). Irradiation doses for personnel at the two commercial VVÉR-440 units are about 20% of the values given above for personnel at the four units. The main contribution to the radiation dose at all units of a nuclear power station results from the performance of repair operations (mainly on equipment in the primary loop).

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*A review of part of the reports at the 3rd All-Union Scientific and Practical Conference on Radiation Safety. The reports are listed at the end of the paper.

TABLE 1. Averaged Values of Adsorbed Activity Q_S and Relative Contribution n_γ to γ Dose from VVÉR-440 at Kol'sk [2]

Nuclide	Q_s, mCi/m²	n_γ, %	Nuclide	Q_s, mCi/m²	n_γ, %
^{110m}Ag	2,6	38,5	^{65}Zn	0,90	2,8
^{60}Co	1,4	19,0	^{95}Zr	0,36	1,4
^{58}Co	3,0	17,0	^{124}Sb	0,11	1,0
^{54}Mn	2,5	12,5	^{140}Ba	0,38	0,40
^{140}La	0,38	4,1	^{103}Ru	0,20	0,20
^{59}Fe	0,45	2,9	^{131}I	0,13	0,20

TABLE 2. Radioisotopic Composition and Distribution of Adsorbed Activity on Equipment Surfaces at Beloyarsk in 1970, % [3]

Unit	^{60}Co	^{54}Mn	^{51}Cr	^{58}Co	^{65}Zn	Total activity, Ci	
						outside core	in core
Unit I	85	6,5	8	0,5	—	345	1200
Unit II	50	5,0	5	—	40	1370	2500

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TABLE 3. Rate of Discharge of Radioactive Aerosols from the Two VVÉR-440 Units at Novovoronezh, mCi/yr [1]

Year	Total β activity	^{131}I	^{137}Cs	^{90}Sr	^{144}Ce	^{95}Zr	^{60}Co	^{54}Mn	^{51}Cr
1972	8,3	1,65	0,17	0,02	0,12	0	0	No measurement	
1973	117,8	65,9	3,5	0,1	0,6	0	0,9	"	"
1974	34,2	5,9	3,2	0,1	1,7	2,6	0,19	"	"
1975	194,9	25,6	15,0	0,3	5,8	3,7	4,3	13,4	11,8
1st quarter 1976	12,6	0,49	1,91	0,01	0,3	0,2	0,2	0,13	0,2
MPD	182000 *	36500	—	365 †	—	—	—	—	—

*All radionuclides with half-lives $T_{1/2} > 1$ day except for ^{131}I and $^{89-90}$Sr.

†Including ^{89}Sr.

The radiation situation at the Kol'sk nuclear power station is even more favorable; in 1974, irradiation of personnel was 0.15 rem/yr and was mainly connected with fuel reloading and the performance of planned preventive maintenance. During nine months in 1975, 0.15 rem out of 0.22 rem resulted from reloading [2].

Units I and II at the Beloyarsk nuclear power station have somewhat poorer characteristics although the mean annual radiation dose for personnel at those units (2.2-3 rem/yr) is below established standards. The main factor in the radiation hazard at these reactors is also the performance of repair operations [3].

At a commercial VVÉR-440 reactor, the calculated specific activity of the primary coolant, assuming that 1% of the fuel elements show gas leakage and 0.1% of the microdefects result in direct contact between nuclear fuel and coolant at the end of a run, is $8 \cdot 10^{-2}$ Ci/liter for the sum of all fission products, which includes $1.6 \cdot 10^{-2}$ Ci/liter from radioactive noble gases (RNG), $5.2 \cdot 10^{-3}$ Ci/liter from iodine isotopes, and $5.8 \cdot 10^{-2}$ Ci/liter from the remaining fission products; there is $4.5 \cdot 10^{-3}$ Ci/liter from corrosion products and 0.12 Ci/liter from "oxygen" activity.

The actual specific activity is 1-2 orders of magnitude less than specified. The short-lived radionuclides ^{18}Fe, ^{24}Na, and ^{42}K at a concentration of $3 \cdot 10^{-4}$ Ci/liter predominate in the water of the primary loop. At Kol'sk, the ^{131}I — ^{135}I content does not exceed $1.5 \cdot 10^{-7}$ to $2.8 \cdot 10^{-4}$ Ci/liter. This is because the number of fuel elements with gas leaks is 30 out of a permissible number of 440 and the number with microdefects is 2 out of a permissible number of 44. There are even fewer defective fuel elements in unit IV at Novovoronezh; the surface contamination of fuel elements with ^{235}U is $6.8 \cdot 10^{-11}$ g/cm^2 as compared to the permissible value of $1 \cdot 10^{-9}$ g/cm^2 [2].

In the neighborhood of the primary loop of a shut-down reactor, the γ-ray field is produced mainly by activated corrosion products adsorbed on the internal surfaces of piping. Typical data for the Kol'sk nuclear power station are shown in Table 1.

The large contribution from ^{110m}Ag is typical; it is formed by activation of stable silver contained in electrodes and certain reactor structures.

At Beloyarsk, ^{60}Co is predominant in deposits on the internal surfaces of the primary loop; its contribution to the dose rate is 97% in unit I and 35% in unit II (Table 2). The main sources of ^{60}Co are Kh13N10T stainless steel and stellite welds in the armature and rods of disconnect devices.

Decontamination performed at unit I of the Beloyarsk nuclear power station led to a reduction in γ-ray levels by a factor of five on the average and by a factor of 25 at particular locations. The effectiveness was less at unit II (a factor of two). Bench tests of a method for high-temperature decontamination gave encouraging results [3].

Commercial VVÉR-440 reactors are recommended from all aspects and particularly from the viewpoint of the rate of discharge of gaseous radioactive wastes. Thus, the annual discharge of RNG from units III and IV at Novovoronezh in 1973-1975 was, respectively 2140, 2056, and 24,785 Ci/yr. Only in 1975 did it reach 2% of the maximum permissible discharge (MPD) of 1.2 million Ci/yr; in the preceding two years it was less than 0.2% [1].

In studies of the degree of dispersion of radioactive aerosols at Novovoronezh, it was established that the radioactivity and mass distributions of aerosols with respect to their aerodynamic diameters were described by a lognormal function. In the stack and in the four ventilating systems, the average aerodynamic

TABLE 4. Air Concentrations of Radionuclides
near the Novovoronezh Nuclear Power Station,
$\times 10^{-17}$ Ci/liter [1]

Year	Point of mea-surement	Total aerosol β-activity	$^{141} + ^{144}$Ce	^{65}Zr	^{137}Cs	^{90}Sr
1972	At power station	9,6	3,8	1,4	0,57	0,27
	60 km from power station	9,3	3,0	1,35	0,32	0,23
1973	At power station	4,0	0,9	—	0,21	0,12
	50 km from power station	3,0	1,0	—	0,21	0,10
1974	At power station	2,0	6,5	4,0	0,40	0,42
	50 km from power station	1,8	5,5	2,8	0,33	0,34
1975	At power station	13,0	11,3	3,9	0,40	0,51
	50 km from power station	10,5	12,1	3,0	0,30	0,55

diameter with respect to particle activity was 2.1-3.9 μ and 0.5-1.1 μ with respect to mass; the average mass concentration was 0.017-0.086 mg/m^3. Precipitation of aerosols in the sampling tube for monitoring aerosol discharges from the large stack of units III and IV at Novovoronezh was 45-60% [4]. Information on the rate of aerosol discharge is presented in Table 3.

Thus, the total 4-yr discharge of radionuclides from two units at the Novovoronezh nuclear power station does not even exceed the mean daily MPD for all isotopes.

At Kol'sk, random leakage of water from the primary loop did not exceed 5 liters/h as compared with a design standard of 200 liters/h. As a result, discharges of RNG during the second half of 1973 came to 87 Ci in all, were 1047 Ci during the whole of 1974, and 430 Ci for four months of 1976. The ^{131}I concentration in air discharged from a stack 120 m high did not exceed $2.5 \cdot 10^{-15}$ Ci/liter, which is 100 times less than the average permissible concentration in air at populated locations. The aerosol components in the discharges were negligible amounts of the short-lived radionuclides ^{88}Rb and ^{138}Cs (0.03 and 0.3 Ci in 1973 and 1974, respectively, [2]).

The radiation environment around the location of the Novovoronezh nuclear power station is mainly characterized by radionuclides of global occurrence (^{90}Sr, ^{137}Cs, ^{144}Ce); "station" nuclides (^{65}Zn, ^{60}Co) make a barely perceptible contribution. Table 4 shows that the air concentration of radionuclides is very low and comparable with the background produced by global fallout. Only a barely perceptible difference is observed between the air concentrations of of nuclides near the nuclear power station and at a monitoring station 50 km away. A similar pattern is also observed in water basins (Table 5).

Calculated radiation doses for the population in the neighborhood of the Novovoronezh nuclear power station are practically the same as the doses resulting from the natural radiation background from global contamination of the biosphere [1].

TABLE 5. Radionuclide Concentrations in
Water (pCi/liter) and Bottom Sediment
(nCi/kg) of the Don River [1]

Year	Medium	Upstream from nuclear power station			Downstream from nuclear power station		
		total β activity	^{137}Cs	^{90}Sr	total β activity	^{137}Cs	^{90}Sr
1972	Water	7,5	0,47	0,58	7,0	0,49	0,54
	Mud	11,4	0,46	0,13	9,3	0,12	0,10
1973	Water	9,4	0,49	0,80	8,8	0,50	0,58
	Mud	10,0	0,11	0,04	11,2	0,15	0,10
1974	Water	10,3	2,6	0,63	10,3	2,45	0,68
	Mud	9,2	0,22	0,10	7,0	0,18	0,14
1975	Water	11,6	1,37	0,46	9,7	1,46	0,52
	Mud	18,3	0,07	0,18	12,9	0,18	0,22

TABLE 6. Radioisotopic Composition of Coolant, Gas, and Deposits in Various Operating Cycles of the BOR-60 Reactor [7]

Operating cycle	Year	Specific activity at coolant, μCi/kg					Surface activity at deposits, μCi/m²					Specific activity of gas, μCi/liter					Dose rate in primary loop boxes during preventive maintenance, μR/sec
		^{22}Na	^{110m}Ag	^{134}Cs	^{137}Cs	^{95}Nb, ^{125}Sb, ^{106}Ru etc.	^{60}Co	^{56}Mn	^{95}Nb	^{140}La	^{137}Cs etc.	^{41}Ar	^{133}Xe	^{135}Xe	^{85}Kr	^{85m}Kr, ^{87}Kr, ^{88}Kr etc.	
Sealed core	1970	50	16	0,2	<0,1	<2	—	—	—	—	—	1,3	$4\cdot10^{-3}$	$2,7\cdot10^{-3}$	—	$\leqslant 1\cdot10^{-3}$	5
Sealed core (with ventilated fuel elements)*	1971	70—190	25—90	~0,3	<0,1	<2	0,2	1	—	—	—	1,3	1—2	0,01—0,1	—	$\leqslant 1\cdot10^{-3}$	20
Operation with defective fuel elements	1972	250—330	110—170	13—160	210	<200	1—7	3—7	40—50	4—170	$\leqslant 10$	1,5	10—100	0,3—5	—	$\leqslant 1$	30—120
	1973	350—500	200—300	200—1300	1800 2000—13000	<400	2—10	5—20	40—160	4—170	$\leqslant 130$	1,5	30—500	1—40	0,3—0,8	$\leqslant 4$	150—300

*Bundle at experimental fuel elements with joints for release of gas after accumulation beneath cladding.

TABLE 7. Total Fission-Product Radio-
activity in the Primary Loop of the BOR-
60 Reactor at the End of Runs, Ci/loop [7]

Run	Nuclide					
	[131]I	[137]Cs	[134]Cs	[136]Cs	[95]Nb	[140]La
Dec. 1971 to Mar. 1972	10,4	5,6	0,36	0,42	0,72	0,07
May to Aug. 1972	280	49	4,3	—	4,1	9,7
Nov. 1972 to Mar. 1973	1100	190	16	15	12	8,7
May to Sept. 1973	190	310	23	18	2,4	1,2

The calculated irradiation of the population in the polar region settlement closest to Kol'sk (12 km) does not exceed 5 μrad/yr with a natural background of 7 μrad/h [2]. The impression is given that the Kol'sk nuclear power station with two-loop VVÉR-440 reactors, which was designed on the basis of the operating experience at Novovoronezh, is the best example of a domestic nuclear power station.

At Beloyarsk, the main sources for RNG discharge are fuel channels with leaky fuel elements and contaminated reactor piles. The existing system for retention of radioactive gases and purification with special filters along with the closed water supply system ensures the cleanliness of the area around the Beloyarsk nuclear power station [3].

At the VK-50 boiling water reactor, which is a prototype for large commercial nuclear power stations with boiling-water reactor vessels, a gas leakage from fuel elements was observed in 1965-1967 that corresponded to 0.5% defective fuel elements. In this case, the coolant activity and gaseous discharges caused no difficulties. On this basis, one can conclude that the presence of 1% of leaky fuel elements does not lead to dangerous working conditions at a nuclear power station [5].

A comparison of the theoretical and experimental data on the specific activities of water, vapor, and condensate showed that the computational techniques satisfactorily described the processes of radionuclide formation and transport in the primary coolant system of the VK-50 reactor. The main components (not considering ^{16}N activity) are isotopes of iodine and RNG, and ^{64}Cu and ^{65}Mn from corrosion products. On the internal surfaces of equipment, 80-90% of the adsorbed activity is caused by ^{65}Zn. The activity of corrosion products in deposits on surfaces inside the core is approximately two order of magnitude greater than that outside it.

In 1972 and 1973, RNG discharges were ~30,000 Ci/yr and 170,000 Ci/yr in 1974 (during normal operation, not more than 1500 Ci/day). To reduce gaseous discharges, tests were made of an experimental adsorption purification system: 20-70 m^3 of SKT brand activated charcoal cooled to —25°C by means of one-stage refrigerators provided a gas cleanup factor of 10^{-2}-10^{-3}.* It was concluded that this method solved the question of fission-product cleanup of gaseous wastes at a nuclear power station with a commercial boiling-water reactor and did not put a limitation on the number of defective fuel elements in the reactor core. From the results of calculations for a 1000 MW(el) commercial boiling-water reactor with 1% of the fuel elements leaking gas, the rate of RNG dicharge in the discharged gases is 3 Ci/sec. Furthermore, the RNG concentration ahead of the turbine will be $2 \cdot 10^{-3}$ Ci/kg, that of ^{131}I, $1 \cdot 10^{-5}$ Ci/kg, and that of other nuclides with $T_{1/2} > 8$ days, $1.5 \cdot 10^{-5}$ Ci/kg [5].

In four years of operation of the BOR-60 experimental fast reactor with sodium coolant, $0.7 \cdot 10^6$ MWh of thermal energy was produced, the equipment was operated for about 17,000 h, and a fuel burnup of better than 10% was achieved. The specific activity of ^{24}Na per MW was 0.7-0.8 Ci/kg of coolant, which led to a γ-ray field of 18 R/h in the neighborhood of piping (at nominal power). Sodium in the secondary loop was practically unactivated. With sealed fuel elements, the main contribution to the specific activity of the primary coolant was made by ^{24}Na, ^{22}Na, and ^{110m}Ag (Table 6). In the opinion of the authors of [7], the practical absence of corrosion products is explained by their low solubility in sodium. The largest deposits of the isotopes ^{140}La, ^{60}Co, ^{95}Nb, and ^{54}Mn were noted in the portion of the piping following the heat exchangers. Radioactivity in the gaseous void of the reactor (≈ 2 m^3) was mainly determined by ^{41}Ar. The constant leakage of

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*According to data in [6], the service life of an absorber filled with 20-40 m^3 of activated charcoal is tentatively 8-10 yr.

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xenon, krypton, and neon corresponded to an average residence time in the coolant of 8 h. Contamination of fuel elements by ^{235}U was close to the permissible level (10^{-9} g/cm^2 of ^{235}U). Discharges of RNG did not exceed 25 Ci/day with sealed fuel elements. In two years of operation, the maximum burnup of fuel elements exceeded 10%, as a rule, at the end of runs and the estimated number of defective fuel elements reached 1%.

In the opinion of these same authors, the activity in the gaseous void does not limit the number of defective fuel elements; it is more important to confine the solid fission products. Although the ^{131}I concentration is comparatively large according to the data in Table 7, it is almost completely removed by the trap for sodium oxides. At the same time, only 30% of the ^{137}Cs is trapped there; from 20 to 40% of its activity can be found in the deposits. Also found in the deposits are ^{140}Ba, ^{140}Le, ^{95}Nb, ^{60}Co, and ^{54}Mn (see Table 6). Since these radionuclides determine the radiation environment in elements of the primary loop after reactor shutdown, drainage of the coolant reduces the γ-ray field insignificantly.

The average individual personnel radiation dose at the BOR-60 installation was 0.3-0.5 rem/yr; no cases of irradiation greater than 2.5 rem/yr were observed in 1971-1973. The contribution of β radiation to the external radiation dose was 2.6%; the maximum energy of the β spectrum did not exceed 1.5 MeV. The appearance of radioactive aerosols (mainly ^{137}Cs and no more than $1 \cdot 10^{-10}$ Ci/liter) was only observed in the central hall during sample collection and during washdown of the reloading shaft with the mean median aerodynamic diameter being 3 and 8 μ, respectively.

Three papers were devoted to the tritium problem. In [8], a theoretical estimate of tritium equilibrium concentration and discharge rate into the environment was made for a nuclear power station with a VVÉR-440 reactor. Activation of deuterium in the coolant, ternary fission of ^{235}U nuclei, and tritium formation from boron in the coolant through the reactions ^{10}B(n, 2α)^{3}H and ^{10}B(n, α)^{7}Li(n, nα)^{3}H were considered. In the absence of random leakage of water from the primary loop, the main source of tritium is the vapor from water-purification basins, in which the tritium concentration in the water can reach $1.8 \cdot 10^{-4}$ Ci/liter. In this case, the tritium is discharged into the environment only along with air and the discharge rate through the stack is ~130 Ci/yr. In cases where the random leakage reaches the design value of 0.2 T/h, the tritium reaches the environment together with debalance water (about 100 Ci/yr) and through the stack (about 30 Ci/yr). In both cases, the tritium concentration in air of working locations does not exceed the mean permissible concentration.

Results of an experimental determination of tritium at a nuclear power station are presented in [9]. Thus, in the water of the primary loop at Novovoronezh, the tritium concentration was the following: 2.2 Ci/liter for unit I, 21 for unit II, 72 for unit III, and 95 for unit IV. The tritium concentrations in the coolant of units I and II at Beloyarsk were 20 and 28 μCi/liter, respectively.

In the review [10], radiation-health evaluation was made of tritium as factor in the contamination of the environment. The total global tritium content increased from the natural level of 20-80 million Ci to 2900 ± 700 million Ci in 1973. While tritium generation was associated predominantly with nuclear explosions up until recently (6-10 million Ci per megaton of thermonuclear explosion), the main source at the present time is factories for reprocessing nuclear fuel (19 Ci/yr per MW of uranium fuel and 36 Ci/yr per MW of plutonium fuel). Apparently tritium will be a critical radionuclide determining the radiation environment in the neighborhood of factories for reprocessing fuel.

An interesting paper is [11], which is devoted to the effect of cooling towers on the dissipation of discharges into the atmosphere from nuclear power stations. There are two mechanisms for changing the aerosol concentration field in the neighborhood of the stack through the effect of a cooling tower. The first is direct washout of contaminants by drops of water which enter the atmosphere from the cooling tower and which are formed by condensation of water vapor. This effect is described in the literature. Since precipitation of the drops occurs close to the cooling tower, it is considered more favorable to locate the stack on the lee side of the cooling tower.

The second mechanism, which has been studied insufficiently thus far, is the local change in turbulent diffusion brought about by the change in atmospheric stratification through the action of thermal discharges from the cooling tower. It was shown by calculation in [11] that for a general wind velocity of 3-7 m/sec, redistribution of contaminant concentration with height occurs in the plume region and the maximum surface concentration is observed closer to the stack, but falloff occurs much more slowly than in the absence of a cooling tower. In the opinion of the authors of [11], therefore, it is more favorable to locate the stack on the windward side of the cooling tower. The relative location of stack and cooling tower, and also the height of the stack, should be taken into consideration in the design of a nuclear power station primarily for the purpose of a sounder determination of the size of the health-protection zone.

The principles for organization of external dosimetric monitoring around the location of a nuclear power station are the concern of [12]. On the basis of data about the radiation environment in the area around the Novovoronezh nuclear power station, which give an idea about the range of propagation of radioactive aerosols, it was concluded that the radius of monitored region should be not less than 10-12 km. The authors proposed two criteria for justification of the volume of dosimetric control: the MPD and existing levels of global fallout. Setting a mean variance of 50-60% from the mean, the authors determined the required number of samples. For example, with the present concentration of β-active aerosols produced by nuclear explosions ($9 \cdot 10^{-17}$ Ci/ liter) and the MPD of 0.5 Ci/day for such aerosols which can produce an air concentration of $2 \cdot 10^{-17}$ Ci/liter out to a radius of 7 km, it is necessary to take 150 air samples per year. Similar methods are also used to determine the volume of monitoring with respect to other parameters. In our opinion, the limitation of such an approach is obvious -- an increase in MPD (and even more, in the actual discharges, which ordinarily do not exceed the MPD) will be accompanied by a reduction in the volume of external dosimetric monitoring, and vice versa.

Several criteria for selection of the location of a nuclear power station are discussed in [13]. In connection with the extensive program for the construction of nuclear power stations (and even more, that for nuclear heat and electric power stations) close to the boundaries of large cities, it is inevitably necessary to analyze the radiation risk to the population with accidental discharges taken into consideration. The fundamental approach to evaluation of the hazard is the determination of the total radiation dose caused by external γ radiation from a passing cloud and by irradiation of the thyroid by radioactive iodine isotopes. The authors present a model for calculation of the total dose for an urban population and an example of its application to a city with a population of $(0.1-5) \cdot 10^6$. It was concluded that technical provision of reactor safety was more efficient than removal of the nuclear power station from the city.

The material presented in these reports confirms the high degree of radiation safety at nuclear power stations for both staff and population; at the same time, they can be useful in the planning of new nuclear power stations.

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RADIATION SAFETY DURING OPERATION OF
HIGH-INTENSITY RADIATION EQUIPMENT*

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UDC 621.039.58.68+541.15

A feature of recent years is the extensive use of high-intensity radiation equipment for the accomplishment of various radiochemical and radiobiological processes on pilot and industrial scales. Among such processes are, in particular, radiation sterilization of medical products, irradiation of agricultural crops to increase their yield, and the production of chemical materials with new or improved properties. The most widely used high-intensity sources of ionizing radiation are ^{60}Co, ^{137}Cs, and electron accelerators with energies up to 10 MeV.

Technical and Organizational Problems in the

Provision of Radiation Safety

Presented in [1] are the results of a study of radiation safety in the radiochemical group at the Institute of Chemical Physics of the Academy of Sciences of the USSR, which has the task of performing fundamental studies of elementary processes in high-energy chemistry, and of the mechanism and kinetics of chemical reactions, and in the development of scientific bases for various radiochemical processes. This group operates the two linear electron accelerators LUÉ-5-2 and U-12-B with accelerated electron energies of 4 MeV and an average beam power of 5 and 0.5 kW, respectively; the group also operates K-200,000 high-intensity radioisotopic equipment, the irradiator of which is assembled from ^{60}Co sources. The methods of radiation protection used for the equipment (concrete operating area with labyrinth entrance), the interlock and warning systems, and organization measures made it possible to provide safe working conditions for personnel. There were no radiation accidents at the installation during 10 years of operation and the radiation level was 1.5-2.5 rem, i.e., 0.3-0.5 of the annual maximum permissible dose (MPD) for irradiation of personnel.

One should note the excessive number of interlocks on the entrance doors to the high-intensity radioisotopic equipment (four), which can be limited to two (for dose rate in the operating area and labyrinth and for closure of the lock on the entrance door to the labyrinth) by replacing the others with appropriate warning signals.

*A review of part of the reports at the 3rd All-Union Scientific and Practical Conference on Radiation Safety. The reports are listed at the end of the paper.

TABLE 1. Parameters of Annual Radiation-Dose Distribution and Probability of Exceeding MPD

Type of operation	Parameters of distrib.		Probability of exceeding dose, %	
	$\bar{D}_g$,	σ	1,5	5
Accident free	0,06 *	1,7	10^{-3}	10^{-5}
	0,18	2,3	0,6	0,1
Loading, adding, replacing sources; repair and maintenance	0,24	3,9	9,5	1,5
	0,39	2,6	9,6	0,7

*First line is for equipment with fixed irradiators, second line for equipment with movable irradiators.
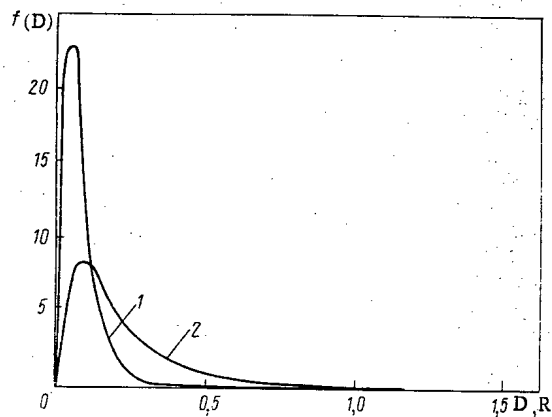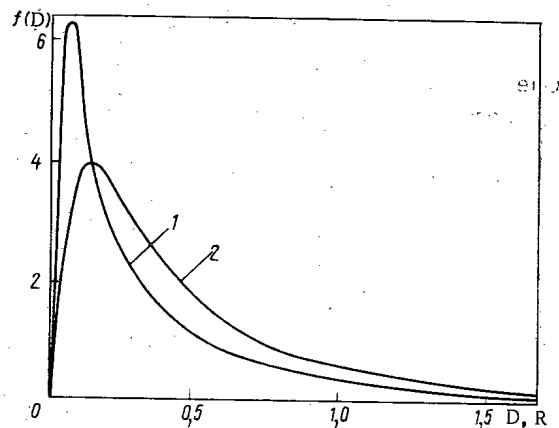
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Fig. 1

Fig. 2

Fig. 1. Annual radiation doses of personnel during accident-free operation of equipment with fixed (1) and movable (2) irradiators.

Fig. 2. Annual radiation doses of personnel during loading, adding, or replacement of radiation sources and during repair and maintenance work (notation same as in Fig. 1).

The products of radiation technology must not be contaminated by radioactive materials because if this is not the case, the residual radioactive contamination of irradiated products used in commerce can become uncontrolled. Because of this, the reliability of the sealing or radioisotopic sources of radiation and its checking during operation of equipment become of overwhelming importance. Under the operating conditions for equipment used in radiation technology, certain factors of a chemical nature have an effect on the individual radioisotopic sources and on the irradiator as a whole along with mechanical effects; these can lead to unsealing of sources with subsequent entrance of radioactive material into the environment.

Based on studies, a rule was proposed [2] for the use of radioisotopic sources in an aqueous medium which took into account the characteristics of radiochemical processes which are accompanied by the release of chlorine and of fluorine-containing materials.

It was shown that the chlorine- or fluorine-ion content of water in reservoirs in the equipment or in other aqueous media must not exceed 0.5 mg/liter at a pH of about 7. The formation of electrochemical pairs should be avoided; in solutions containing chlorides, it is necessary to use stainless steel with an increased content of nickel or titanium (up to 20-40%).

A method was proposed for decontamination and rejection of ^{60}Co and ^{137}Cs sources based on successive immersions for two minutes in two or more volumes of a 5% solution of oxalic acid.

Noting the practical value of the results obtained, it would also be worthwhile to propose a scheme for checking ^{60}Co and ^{137}Cs sources not only under conditions where there are shielded rooms in the installation complex, but also under conditions when there are none since the activity of a single radioisotopic source amounts to 1000 Ci and more.

One should note a carefully formulated experiment to study the solubility of ^{60}Co in aqueous media in high-intensity radioisotopic equipment. From data in the literature, it is known that the solubility of ^{60}Co in water is from 10^{-7} to 10^{-4} g/cm^2-day. However, these values are not acceptable for making the proper decisions about the creation of cleaning equipment and the collection of radioactive wastes since their use is associated with considerable cost. The relationship obtained for the dependence of the solubility K of ^{60}Co on the temperature T of the water in which a ^{60}Co source is operating was $K(T) = 1.3 \cdot 10^{-5} (1 + 0.0168T)$. For practical purposes, one can assume the solubility is $2.2 \cdot 10^{-5}$ g/cm^2-day.

Organizational measures directed toward the provision of radiation safety during operation of equipment with electron accelerators in local biological shielding are discussed in the operating instructions described in [3].

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Radiation Levels for Personnel Operating

High-Intensity Radiation Equipment

One of the indices of operating safety at high-intensity radiation equipment is the level of the individual radiation dose for personnel loading, adding, and replacing radioisotopic radiation sources, and for personnel operating such equipment.

The data presented in [4] are evidence of the effectiveness of the radiation-protection system in the "Sterilizator" radioisotopic gamma equipment used in the medical supply industry and in agriculture. During the operating period of this equipment, external radiation levels for the personnel did not exceed permissible values and radioactive contamination resulting from ^{60}Co was only observed in trace amounts.

The high quality of the design and construction of the equipment and the effectiveness of its shielding makes it possible at the present time to do without certain forms of monitoring previously considered necessary. There then arises the problem of organizing an optimal system of radiation monitoring at installations which makes possible a reduction in the cost of the monitoring and an increase in the economic efficiency of high-intensity radiation equipment.

A sound solution of the problem of organizing radiation monitoring is impossible without an analysis of the radiation levels for personnel employed in the operation of high-intensity radioisotopic equipment. Such an analysis was made [5] on the basis of statistical data on methods of radiation monitoring and on annual personnel radiation doses. Individual radiation doses were analyzed for two groups of equipment (with movable and fixed irradiators) differing significantly in construction and irradiator activity.

The value of the mean annual dose over the period of equipment operation was chosen for analysis. It can be assumed that this quantity is distributed in accordance with some probability law because of the large number of independent factors affecting it so that the analysis of the data obtained employed the methods of mathematical statistics. Statistical analysis of the data showed that the distribution of the mean annual doses received by personnel at the two groups of equipment when performing various types of operations is described rather well by a lognormal law (Figs. 1, 2). The known shape and parameters (mean geometric dose $\bar{D}_g$, R, and the standard deviation σ) of the radiation dose distributions provide an opportunity to determine the probability of exceeding any given annual dose (see Table 1). The data shown make it clear that for accident-free operation of equipment, the individual doses received by personnel practically do not exceed 0.3 of the annual MPD. According to the requirements of the Radiation Safety Standards (NRB-69), individual dosimetric monitoring and special medical examination are not performed for people whose working conditions are such that the radiation dose is systematically less than 0.3 of the annual MPD. At the same time, radiation doses of personnel may exceed maximum permissible values when loading, adding, or replacing radiation sources, or when performing repair and maintenance work so that individual dosimetric monitoring is necessary in those cases.

Considering that the activity of irradiators can amount to 10^6 Ci and more in a single installation, service personnel should wear accident dosimeters at all times.

Overall Estimate of Working Conditions for

Personnel Employed in Radiation Technology

The realization of several radiation —technology processes on commercial scales poses an urgent problem in the overall health evaluation of the influence of factors in the industrial environment on the health of personnel.

The solution of this problem was advanced in two fields in [6]:

the determination of the most probable types of personnel contact with factors having an adverse effect under industrial use of high-intensity radioisotopic gamma equipment;

evaluation of the dependence of the harmful effects of γ radiation on the constantly associated nonradiation hazard factors — the products of radiolysis of air (ozone and oxides of nitrogen).

It was established that during accident-free operation of high-intensity radioisotopic equipment, products of the radiolysis of air (ozone and oxides of nitrogen in the ration 1 : 4) are always formed in the air of the compartment in which the equipment is operating in a concentration that exceeds the maximum permissible concentration (MPC) by factors of 60-100 in some cases.

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It was shown that in an accident at an installation, personnel can be subjected to various types of combined effects from ionizing radiation (in doses up to several hundreds and thousands of rem) and from the products of radiolysis of air in concentrations exceeding the MPC, which can represent a serious hazard even in the absence of radiation effects. This should be taken into consideration in the organization of safety and rescue operations.

Experimental data were presented [6] on the combined effect of γ radiation and of the products of radiolysis of air in concentrations corresponding to accident situations. In addition to integral toxicological evaluation (survival rate, mean lifetime of experimental animals) and indicators of a pathological nature (spleen mass factor), studies were also made of radiosensitive processes such as resynthesis of macroergic compounds and phosphorus-containing albumens in the tissue of the spleen. It was shown that introduction into the irradiated medium (independently of irradiation dose) of products of the radiolysis of air (less than 10 MPC in terms of ozone) had no effect on the course and outcome of radiation injury. Further increase in the concentration of these products (up to 25-50 MPC) led to a marked reduction in the seriousness of radiation injury.

The material presented in these reports is evidence of the fact that the fundamental problems of safety in radiation technology are being successfully solved. Further studies should be carried out which are directed toward optimization of safety systems in general and toward optimization of radiation monitoring systems in particular. The results of such studies will have an effect on the economic characteristics of radiation technology. It is also necessary to continue the study of the overall effect of hazard factors of both radiation and nonradiation nature on the working conditions of personnel employed in radiation technology including the radiotoxicological experiments on the joint effect of these hazard factors of living organisms.

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THE RELATIVE DANGER OF NUCLEAR POWER STATIONS (NPSs)
AND THERMAL POWER STATIONS (TPSs) FOR THE ENVIRONMENT

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At the end of 1975 the total installed power of Soviet power stations amounted to 218,000 MW. A further considerable increase of power is being planned for nuclear power generation. The principal reactor types are the VVÉR and RBMK series. At present, the unit capacities of these reactors, achieved and to be achieved, amount to 440-1500 MW. However, up to now thermal power stations (TPSs) comprise the basis of the country's power generation. Because of the continuing construction of TPSs it will be interesting to compare the ecological effect of TPSs and NPSs of identical electrical capacity.

The thermal discharge from NPSs at present is higher than from TPSs. This is because the efficiency of a NPS amounts to 30-32% in comparison with the 35-40% which is characteristic for TPSs with a conventional steam — water cycle. However, nuclear power generation has greater unused reserves for increasing the thermal efficiency: the discharge of heat from a NPS into the reservoir-condensers could be even less than conventional TPSs. It is true that this refers predominantly to the future high-temperature gas-cooled and fast sodium-cooled reactors, but even according to the most optimistic estimates their fraction of the total nuclear power generation capacity up to the end of the century will not exceed 30% and, consequently, nuclear power generation remains predominantly thermal. The amount of water required by NPSs with thermal reactors, and its reduction with time are shown in Table 1 [1].

The economically most profitable straight-through cooling, because of the shortage of fresh water, is being superseded by recirculation. The latter, however, requires either the confiscation of large territories [e.g., for cooling a NPS with a capacity of 1000 MW(el), a reservoir with an area of 5-7 km^2 is required], or the construction of very expensive cooling towers. The decrease of efficiency and increase of capital costs, due to fulfilling the demands for "thermal protection" of the environment, can be seen from the data of Tables 2 and 3 [1]. The transition to the construction of NPSs with recirculated water supply and the use of cooling-towers ensure adequate protection of the water resources of the environment from thermal effluents. At the same time, it should be noted that the problem of the joint effect of thermal and radioactive effluent from NPSs requires a detailed study both in the running of carefully controlled laboratory experiments, and also by means of observations in the vicinity of operating nuclear power stations.

The ecological advantages of NPSs over TPSs are manifested in the analysis of gaseous and aerosol effluents and their risks to the environment. First and foremost, in contrast from TPSs, the operation of NPSs is not associated with the burning of fuel, the oxygen content in the atmosphere is not reduced, and the

TABLE 1. Specific Consumption of Cooling Water, liter/(kW · h)

Year	NPS	TPS	TEPS
1970	220	132	113
1975	200	127	101
1980	160	120	80
1985	150	118	67
1990	138	110	56
2000	125	104	50

TABLE 2. Characteristics of Power Station Cooling Systems

Characteristic	Straight through	Recirculation	
		with water reservoir	with cooling towers
Av.yearly temp. of cooling water, °C	11	15	22
Condensation temp., °C	24	30	40
Pressure in condensers, kg/cm^2	0,030	0,043	0,075
Efficiency, %	38	37	34
Specific capital investments, rel. units	1,0	1,6	2,3

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TABLE 3. Power Station Cooling System Used

Years	Straight through	Recirculation	
		with water reservoir	with cooling towers
Up to 1950	70	20	10
1955–1960	50	38	12
1965—1970	40	43	17

concentration of carbon monoxide gas is not increased. NPSs do not change the chemical composition of the atmosphere, the hydrosphere, or the soil. For a quantitative comparison of the effect of gaseous and aerosol effluents from NPSs and TPSs on the environment, an indirect method must be chosen — to calculate and compare the amount of air necessary for reducing the content of harmful impurities to limiting permissible concentrations (LPC) for chemical, and average yearly permissible concentrations (APC) for radioactive substances. It follows from Table 4 that a TPS, in comparison with a NPS (with a PWR reactor) of the same electrical capacity, contaminates volumes of air greater by a factor of hundreds and thousands [2].

Estimates of the risks from gaseous effluents of NPSs and TPSs lead to a similar conclusion, by the method of comparing the effluent output with the limiting permissible burdens (LPB) of harmful substances for the human organism. Gaseous and aerosol effluents from coal- and oil-fired TPSs (Table 5) are several thousands of times more dangerous than the gaseous effluents from NPSs. The contribution of microelements contained in burnt organic fuel (including predominantly vanadium, nickel and lead), can be compared with the harmful effect of the principal contaminants (SO_2, nitrogen oxides) which are characteristic for the operation of TPSs [3].

It is well known that natural radioactive substances are dispersed over the entire medium surrounding mankind. They are contained also in organic fuel burned by TPSs. By taking into account this factor, one is led to the conclusion that the radiation hazard from TPSs is higher than from NPSs. The results of these measurements and calculations were presented by Polish scientists at the recent (1973) IAEA Conference. Thus, the radiation dose intensity to the skeleton, due to ^{226}Ra discharged from TPSs in the combustion products, is equal to 0.1 mrem/yr per MW (el). The dose due to operation of NPSs amounts to $0.2 \cdot 10^{-3}$ mrem/yr for the whole body and 10^{-7} mrem/yr for the thyroid. Thus, the radiation effect dosage from TPSs is a minimum of a factor of 500 greater than during the normal operation of NPSs.

The exceptionally low figures for the additional irradiation of the population in the vicinity of NPSs prompt, in certain cases, the elimination of one or other of the protective barriers. For example, in boiling nuclear reactors, in order to increase the efficiency, the steam formed in the primary circuit is fed directly to the turbine. As a result, the discharge of gaseous radioactive substances into the environment increases in these NPSs. The published results of the comparison between pressurized water reactors (PWR) and boiling water reactors (BWR) are shown in Table 6, and confirm that in NPSs with BWR, the activity of the gaseous effluents is greater than in NPSs with PWR. Nevertheless, the "margin" up to the dose limit is sufficiently large that NPSs with BWR also can be assumed to be radiation-safe for the population. For these NPSs, the maximum yearly radiation dosage to the population (1 mrem) is less by a factor of three than the radiation effect to which a man is subjected, on the average, during a single flight in a modern jet aircraft (3 mrem).

TABLE 4. Volume of Air Required to Reduce the Content of Harmful Effluents from NPSs and TPSs with a Capacity of 1000 MW (el) to the LPC [2]

Type of power station	Impurity	Output of effluent	LPC or APC	Required volume $\times 10^9$ m^3
Coal-fired TPS	SO_2 (3,5% S)	$1,4 \cdot 10^8$ kg/yr	80 µg/m^3	$1,8 \cdot 10^6$
	Aerosols (ash content 15%)	$4,5 \cdot 10^6$ »	75 »	$6,0 \cdot 10^4$
Oil-fired TPS	SO_2 (1,6% S)	$0,52 \cdot 10^8$ »	80 »	$6,6 \cdot 10^5$
	Aerosols (ash content 0.05%)	$0,72 \cdot 10^6$ »	75 »	$9,7 \cdot 10^3$
Gas-fired TPS	SO_2	$1,36 \cdot 10^4$ »	80 »	170
	Aerosols	$0,45 \cdot 10^6$ »	75 »	$6,0 \cdot 10^3$
NPS with PWR	Radionuclides of inert gases	$1,6 \cdot 10^4$ Ci/yr	$3 \cdot 10^5$ pCi/m^3	55
	^{131}I	0,15 »	100 »	1,5

TABLE 5. Relative Hazard of Gaseous Effluents from TPSs and NPSs for the Human Organism, in Numbers of LPB

Type of power station	Principal contaminants	Microelements
Coal-fired TPS	2,01	2,73
Oil-fired TPS	1,69	0,64
NPS with PWR	0,002	0,001

TABLE 6. Discharge of Gaseous Radioactive Substances from a NPS with BWR and PWR Reactors

Characteristic	Type of reactor	
	BWR	PWR
Radioactive noble gases (RNG):	Short-lived isotopes, radioact. noble gases	$^{85}Kr + ^{135}Xe$
Output of effluent, Ci/yr	$1,33 \cdot 10^6$	$16 \cdot 10^6$
APC, pCi/m^3	$3 \cdot 10^4$	$3 \cdot 10^5$
Vol. of air for dilution to the APC, $\cdot 10^9 \ m^3$	$44 \cdot 10^3$	55
^{131}I:		
Output of effluent, Ci/yr	6,6	0,15
APC, pCi/m^3	100	100
Vol. of air for dilution to APC, $\cdot 10^9 \ m^3$	66	1,5

The results of the assessment of the radiation effects of NPSs (during normal operation) on the population, according to data of the National Agency for the Preservation of the Environment of the USA, are thus: natural radiation background 100-150 mrem/yr; medical procedures (x-ray diagnostics, etc.) 100 mrem/yr; radioactive fallout 5 mrem/yr and NPSs 0.01 mrem/yr.

In the case of an accident, the radiation environment may be considerably worsened. This forces one to stipulate additional technical measures for the prevention of accidents and for the reduction of their consequences. This is concerned first and foremost with increasing the reliability of the entire reactor plant, and also the presence of an emergency cooling system for the reactor core, protective devices for containing fission fragments released from the fuel, etc. For example, in the Novovoronezh NPS, in cases of a large leak from the primary circuit (more than 100 tons/h) a system of accident localization has been provided, which excludes the ejection of radioactive coolant outside of the special leak-tight compartments calculated on the elevated pressure in them. In addition to the automatic disconnection of the leaktight compartments from the air-conditioning, sprinkler (spraying) installations are brought into operation simultaneously, for condensing the steam and thus reducing the pressure in the pipeline compartments.

A theoretical analysis of the possible consequences of a radiation accident in a NPS with a typical VVÉR-440 unit leads to the following conclusions: in the case of a hazard associated with the loss of coolant without breakdown of the hermiticity of the fuel elements, no excessive radiation hazard arises for individuals, even as a result of operation during 200 effective days. This same conclusion is substantiated for an emergency situation when there is 1% of nonleaktight fuel elements in the active zone. In the case of an accident with partial meltdown of the core (with fuel of sintered uranium dioxide), the maximum dose values do not exceed tens of rem to the thyroid gland.

Without considering this vast problem in more detail, we note only the principal results:

1) During operating experience of more than 1000 reactor-years in nuclear power generation, no accident has occurred with injury to the population or other components of the biosphere;

2) Modern technology of reactor construction has achieved such reliability that the probability of destruction of NPS from an accident will be found to be less than the result of the effect of such elemental factors as lightning strikes or earthquakes, and is comparable only with the probability of the impact of a large meteorite. This risk is less by a factor of 100, approximately, than the risk of being maimed as a result of an automobile catastrophe.

At the 4th International Conference on the Peaceful Uses of Nuclear Energy (Geneva, 1971), the results were presented of calculations carried out by Soviet scientists of the ecological consequences of the development of peaceful nuclear power generation, based on the utilization of conventional types of organic fuel and fissile materials. The calculations showed that by using petroleum and gas, by the year 2000 A.D. a constant tenfold excess of the average daily LPC for sulfur dioxide and fly-ash will occur in the air everywhere above the land. With this same scale of utilization of nuclear power generation, the content of radioactive substances

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in the surface layer of the atmosphere will not exceed 1/1000 of the APC. Thus, nuclear power generation contaminates the outside environment to a factor of 10,000 less than conventional power generation.

Criteria for the permissible content of radioactive substances in the environment have been produced on the basis of an exceptionally rigid approach, to which conventional toxicology and communal hygiene is only just starting to approach. Radiobiologists and specialists in the field of radiation hygiene, in support of the APC values, have started from the supposition that the air or water containing the radionuclides are required by mankind continuously and at a constant rate. As the APC, they take the concentration when the internal radiation dosage created by the equilibrium activity of the accumulated radionuclide does not exceed the permissible values. This computation of the cumulative effect of harmful substances still has not found widespread application in hygiene — as a rule, the LPC of toxic substances have been established on the basis of short-time experiments and have a low safety factor up to clearly harmful threshold concentrations.

A direct comparison of the ecological consequences of the development of world power generation of the two types convincingly corroborates in favor of nuclear reactors and NPSs. The widespread construction of NPSs is one of the most ideal means of protecting the atmosphere from harmful industrial effluents. However, in order to assess completely the ecological effect of the various methods of electric power generation, it is necessary to look at the problem more broadly and to analyze the whole of the fuel cycle. In fact, a NPS represents only a small part of this complex cycle, scattered over the territories of the country, in contrast, for example, to a geothermal thermal-electric power station, where all the elements of the cycle are concentrated in one place.

According to the graphic expression of the American ecologist P. J. Bowen, a NPS is similar to the tip of the iceberg of nuclear power generation. Of the three principal components of the fuel cycle — production of fuel, electric power and waste disposal — we have considered briefly only the middle component. An ecological analysis of the entire fuel cycle for nuclear power generation is a most important problem for the immediate future.

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DEPOSITED PAPERS

THE POSSIBILITIES OF DETERMINING COPPER TRICHLORPHENOLATE
IN PLANTS BY THE NEUTRON ACTIVATION METHOD*

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The problem is posed of assessing the feasibility of radioactivation analysis to determine the residual quantity of copper-containing pesticides (by the example of copper trichlorphenolate — CuTCP). Taking into account the technique of treating cotton seeds with CuTCP and the average content of copper in the plants, it can be assumed that in order to determine CuTPC it is necessary to carry out a preliminary extraction from the sample being analyzed.

The extraction of CuTCP from cotton seeds was carried out with dichlorethane or carbon tetrachloride. After concentration of the extract to a very small volume, the residue is transferred completely to a polyethylene boat. The packed samples, together with standards, are irradiated with a flux of $1 \cdot 10^{13}$ n/cm$^2 \cdot$ sec over 1 h and after 15 h of "cooling," the γ spectra are measured. The copper content is found by comparing the height of the 0.51-MeV photopeaks in the sample and standard. The reliability of the procedure has been verified on simulated samples with known content of CuTCP.

It has been established that with a low content of copper trichlorphenolate also in the blank experiment, in the region of the copper γ emission (0.51 MeV) an interfering effect of ^{24}Na is detected, which is taken into consideration by deducting the normalized spectrum of the sodium standard. Analysis of the untreated seeds showed that the total of the errors in the blank experiment (including also those due to extracted forms of copper) does not exceed n $\cdot 10^{-8}$ g per sample. Consequently, the limit of detection may be assumed to be $1 \cdot 10^{-7}$ g of copper or $7 \cdot 10^{-7}$ g CuTCP in the sample.

If we consider that the average content of copper in the polyethylene amounts to 10^{-6}-10^{-7}%, then under conditions of the selective separation of the copper occurring in the composition of CuTCP, the sensitivity can be increased to ~0.01 μg per sample. Thus, the high sensitivity, simplicity, and high potential reproducibility of the method allows one to assume it to be promising for the determination of copper trichlorphenolate and, obviously, of other copper-containing pesticides.

* (No. 871/8458. Paper submitted August 6, 1975, summary May 13, 1976. Total text 0.2 author folio, 1 table, 5 references).

PULSED ELECTRON CURRENT EXCITED

BY γ RADIATION IN AIR*

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The excitation of an electron current in air by pulsed γ radiation with an energy between 10 keV and 10 MeV was considered. The considerations included Compton scattering, photoabsorption, and the formation of electron – positron pairs.

A general formula was obtained for the electron current $j_\delta (\varepsilon, t)$ excited by a short pulse (δ pulse) of monoenergetic γ radiation of energy ε, with the radiation propagating in the form of a plane front in a medium. The approximation of independent energy transfer events and electron scattering [1] was used to determine the velocity v'(t) for the ordered motion of a fast electron; the velocity v'(t) appears in the formula for $j_\delta (\varepsilon, t)$. It was shown that electron scattering can cause a substantial reduction of the velocity v'(t) of the ordered motion relative to the intrinsic velocity of an electron which is being slowed down. Based on the expression which was found for v'(t), the pulsed electron current excited by instantaneous γ radiation was determined by numerical integration. Figure 1 shows the functions $j_\delta (\varepsilon, t)$ in units of τ^{-1} for air ($\tau = ct/l$ denotes the dimensionless time; c denotes the velocity of light; l denotes the total shift of an electron with the initial energy of 1 MeV; according to the calculations, the shift is 2.04 m for air with normal density). The pulsed electron current which was calculated with a simple model of the moderation of a fast electron without scattering [2] (curve 1) differs strongly from the results of the present work (curve 2) (the results of [2] are 2-3 times greater).

The results for the total shift of a fast electron are compared with the results of experiments on the passage of fast monoenergetic electrons through metal foils. A formula for the electron current excited by quasi-stationary fields of γ quanta was studied.
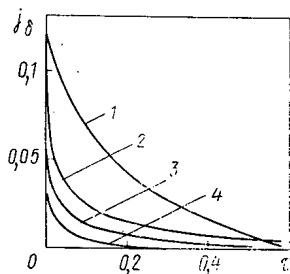

Fig. 1. Pulsed electron current excited by instantaneous γ radiation in air at ε = 2 MeV: 1) without scattering; 2, 3, 4) with scattering at ε = 2, 1, and 0.5 MeV, respectively.

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LETTERS

RATE OF CHANGE OF REACTIVITY OF THE BOR-60
REACTOR DURING THE OPERATING PERIOD

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During the six years of operation of the BOR-60 reactor, extensive investigations and regular monitoring have been carried out [1] of the changes of reactivity of the active zone.

When planning the next fuel rechargings, leading to a change of composition and size of the active zone, it is important to know in advance how they will affect the reactivity reserve of the system and its rate of change during operation of the reactor.

The reserve of reactivity in the BOR-60 reactor has been measured for a different number of fuel assemblies with different burnups. For this, the average burnup did not exceed 5.5% because of the replenishment with fresh fuel. Figure 1 shows the rate of change of reactivity $\Delta\rho/\Delta W$ (ρ is the reserve of reactivity, W is the energy produced).

As any changes taking place in the active zone are reflected on the magnitude of the reactivity, results are given in this present paper obtained for operating conditions of the reactor at the minimally controllable power level. In this case, the reactivity power effect has no effect on the rate of change of reactivity, which will be determined in this case mainly by the changes of the isotopic composition and dimensions of the active zone.

In order to analyze the experiments and the feasibility of predicting the duration of operation of the reactor after the next recharging, the change of reactivity as a function of the energy produced and the charge of ^{235}U was determined by reactor calculations with an active zone and shields close to the constitution of the BOR-60 reactor, on a BÉSM-4 computer in few-group diffusion approximation in two-dimensional geometry according to the program in [2]. The BNAB-70 (British Nuclear Data Library) constants served as the the input data for averaging. The burnup was calculated for active zones with fresh and with partially burned-up fuel, and in this case the dimensions of the active zone and, correspondingly, the charge were increased in order to achieve criticality of the initial state. The results of the calculations of the rate of change of isotopic composition of the fuel during burnup as a function of the fuel charge are shown in Fig. 1 (curve 1). The nature of this dependence is quite obvious, i.e., with increase of fuel charge the rate of decrease of the reactivity reserve is reduced, and for a small range of change of ^{235}U charge, the dependence is almost linear.
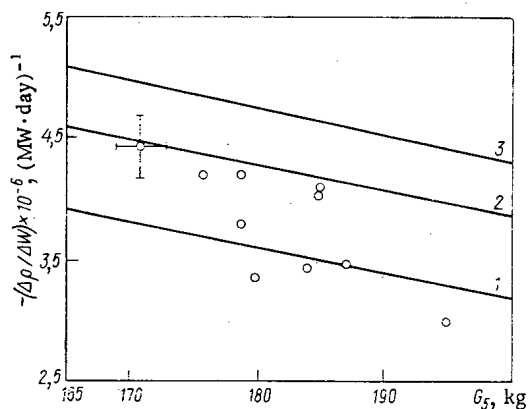

Fig. 1. Dependence of rate of change of reactivity reserve in the BOR-60 reactor on the ^{235}U charge in the active zone; O) results of measurement on the reactor; ———) calculation.

It is well known [3] that during operation of a reactor, swelling of the fuel takes place by the action of neutron irradiation. This affects the change of reactivity of the system as, in consequence of the radiation swelling, the fuel density is reduced and the height of the active zone increases. In particular, because of axial swelling, the nuclear concentration of the fuel is reduced by a factor of $1 + \Delta h/h$, where h and Δh are the height of the active zone and its increase as a result of swelling.

The results of the calculations, carried out on the assumption that the rate of volume radiation swelling during burnup corresponds to 0.55 and 1% per 1 at. % of fuel burnup, are shown in Fig. 1 by curves 2 and 3.

The rate of change of reactivity is increased in absolute magnitude due to taking account of the contribution of the axial swelling of the fuel, but the nature of its dependence on the ^{235}U charge remains almost unchanged.

The experimental data are disposed mainly between curves 1 and 2, i.e., the contribution of swelling for all charges amounts to 0-0.55% per at. % fuel burnup.

Data are contained in [4] which show that swelling of uranium dioxide during irradiation does not take place uniformly but is of a threshold nature: up to a burnup of approximately 5.7% of heavy atoms, it swells at a rate of 0.5% per at. % of uranium burnup, and then at a rate of 1.7% per at. % of uranium burnup.

The active zone of the BOR-60 reactor contains fuel with a relatively low burnup (< 5%) and a burnup of up to 12%. At different times of operation, 30 to 50% of fuel bundles have been found in the active zone of the reactor with a burnup > 5%. If we consider that after reaching a burnup of ~5.5% the rate of radiation swelling of the uranium dioxide increases significantly, then the experimental results are found to be difficult to explain. It is known, however, that swelling of the fuel leads to its adhesion with the cladding. Material-study investigations of the fuel elements withdrawn from the BOR-60 reactor have shown that this occurs on reaching a burnup of 5-6%.

On further irradiation of the fuel its swelling, particularly in the axial direction, will exert a considerable stress on the fuel element cladding. In consequence of this, although with an increase of the radiation dosage there is a tendency to increase the rate of swelling of the fuel, the rate of swelling in the axial direction because of the resistance of the fuel element cladding may remain at the previous level, or even be reduced, which obviously also explains the results obtained.

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A METHOD OF DETERMINING THE THERMOPHYSICAL
PROPERTIES OF REACTOR MATERIALS AT
ELEVATED TEMPERATURES

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UDC 536.21.:621.039.54

The determination of the thermophysical characteristics of reactor materials and, in its turn, of the nuclear fuel, has its special features, as it is desirable to use small samples of simple geometrical shape (e.g., pellets). Because of this, it is advantageous to use the method of a thin wafer heated up from one side stepwise and with a varying thermal flux.

An equation was obtained in [1] which describes the change of the steady-state temperature in a plate of thickness δ by a quantity ϑ at a point with the coordinate x at the instant τ after a change of the thermal flux density heating the plate with a surface area $x = \delta$, by an amount ΔH_s, with the condition that the heat exchange with the surface is described by Newton's law

$$\vartheta\,(x\mathrm{Fo}) = \frac{\Delta H_s \delta}{\lambda} \left[\frac{1 - \dfrac{\mathrm{Bi}_0}{\delta} X}{\mathrm{Bi} - \dfrac{R\,(\mathrm{Bi})^2}{(1-R)^2}} - 2\sum_{n=0}^{\infty} \frac{\cos\, y_n \dfrac{x}{\delta} + \dfrac{\mathrm{Bi}}{y_n\,(1+R)} \sin\left(y_n \dfrac{x}{\delta}\right)}{D_n \exp\,(y_n^2 \mathrm{Fo})} \right], \tag{1}$$

where

$$D_n = y_n \sin y_n \left[1 + \mathrm{Bi} - \frac{2\mathrm{Bi}_0\mathrm{Bi}_\delta}{\mathrm{Bi}} + \frac{y_n^2}{\mathrm{Bi}} + \frac{\mathrm{Bi}_0\mathrm{Bi}_\delta}{y_n^2} + \frac{(\mathrm{Bi}_0\mathrm{Bi}_\delta)^2}{\mathrm{Bi}\,y_n} \right];$$
$$R = \mathrm{Bi}_0/\mathrm{Bi}_\delta; \quad \mathrm{Fo} = a\tau/\delta^2.$$

Here y_n is the root of the equation $\cot y = (y/\mathrm{Bi}) - (\mathrm{Bi}_0\mathrm{Bi}_\delta/\mathrm{Bi})(1/y)$; a and λ are the thermal diffusivity and thermal conductivity of the plate material, respectively;

$$\mathrm{Bi} = \mathrm{Bi}_0 + \mathrm{Bi}_\delta, \quad \mathrm{Bi}_0 = \alpha_0\delta/\lambda, \quad \mathrm{Bi}_\delta = \alpha_\delta\delta/\lambda,$$

where α_0 and α_δ are the heat-transfer coefficients from the surface $x = 0$ and $x = \delta$, respectively.

An attempt [1] to use the equation obtained for the experimental determination of the thermal diffusivity of a plate by direct extrapolation of the linear (on a semilog scale) part of the temperature change with time showed that this procedure is totally unsuitable for relatively good heat conductors, e.g., metals and metal-like compounds.

However, a new method can be suggested for determining a set of thermophysical characteristics of materials during a stepwise change of the thermal flux heating up the sample being studied — a plate. Differentiating expression (1) with respect to Fo when $x = 0$, we obtain

$$\frac{\partial\vartheta}{\partial\mathrm{Fo}} = 2\frac{\Delta H_s \delta}{\lambda} \sum_{n=0}^{\infty} \frac{y_n^2}{D_n \exp\,(y_n^2 \mathrm{Fo})}. \tag{2}$$

For fixed values of R and Bi, the function $\partial\vartheta/\partial\mathrm{Fo}$ has a maximum for a certain value of $\mathrm{Fo} = (\mathrm{Fo})_{max}$. The estimates made showed that with not too large values of $\mathrm{Bi}\,(\mathrm{Bi} \leq 0.3)$, this maximum is very "sluggish," i.e.,
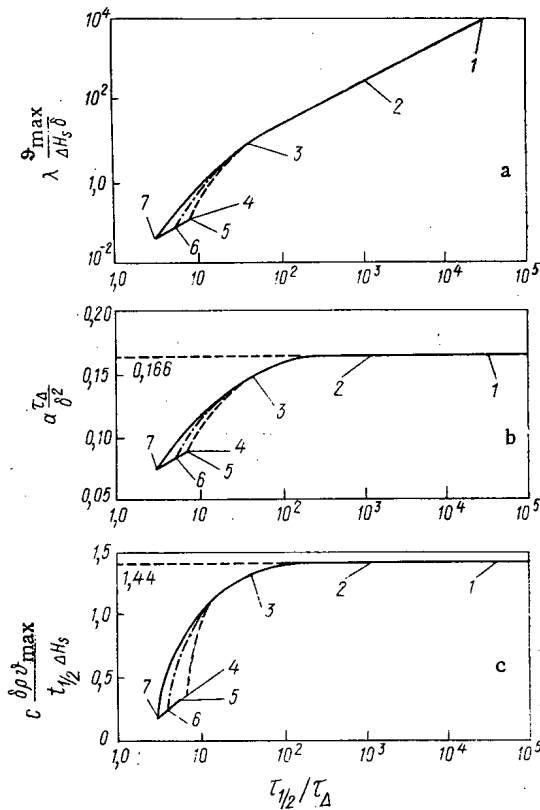
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Fig. 1   Fig. 2

Fig. 1. Graphs for determining the thermal conductivity (a), thermal diffusivity (b), and specific heat (c) of the plate material: 1) $Bi = 10^4$; 2) 10^{-2}; 3) 1; 4) 10; 5) $R = 10^7$; 6) $R = 5$; 7) $R = 1$.
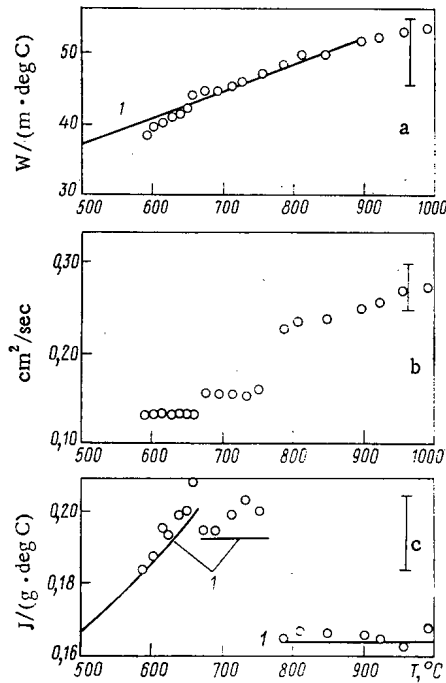
Fig. 2. Dependence of thermal conductivity (a), thermal diffusivity (b), and specific heat (c) of cast uranium on temperature [O, 1) data of this paper and of [2], respectively].

over considerable surroundings of the point $(Fo)_{max}$ the derivative $\partial \vartheta / \partial Fo$ coincided with $\partial \vartheta [(Fo)_{max}] / \partial Fo$ with a high degree of accuracy. This means that there exists a considerable time interval in which ϑ is varying almost linearly.

Thus, the experimental possibility exists of determining quite accurately the intercept τ_Δ, cutting the time axis of the tangent, drawn with maximum slope to the curve $\vartheta(\tau)$. For three fixed values of R, the generalized time $(Fo)_\Delta$ was calculated as a function of the quantity Bi:

$$(Fo)_\Delta = (Fo)_{max} - \vartheta [(Fo)_{max}] \frac{1}{\frac{\partial \vartheta}{\partial Fo} [(Fo)_{max})]}. \tag{3}$$

With the lapse of a sufficiently long time (Fo→∞), the temperature at the surface $x = 0$ approaches the steady-state temperature

$$\vartheta_{max} \simeq \frac{\Delta H_s \delta}{\lambda \left[Bi + \frac{R(Bi)^2}{(1+R)^2} \right]}. \tag{4}$$

For the very same fixed values of R, the behavior of the generalized time $(Fo)_{1/2}$ necessary for heating up the surface $x = 0$ up to one-half of the maximum temperature ϑ_{max} was calculated. By constructing the ratio $(Fo)_{1/2}/(Fo)_\Delta$ for fixed values of R and with different values of Bi, a single-valued relation can be obtained between $(Fo)_{1/2}/(Fo)_\Delta$ and Bi. By substituting λ in formula (4) by the thermal diffusivity, the specific heat C and the density of the material ρ, and by expressing the thermal diffusivity in terms of $(Fo)_{1/2}$ and $\tau_{1/2}$, we obtain an expression defining the specific heat of the plate material:

$$ C \frac{\delta \rho \vartheta \max}{\tau_{1/2 \Delta} H_s)} = \frac{1}{(Fo)_{1/2} \left[Bi + \frac{R \, (Bi)^2}{(1+R)^2} \right]} . $$

Taking into account the conformity between $(Fo)_{1/2}/(Fo)_{\Delta}$ and Bi, the calculated relation for determining the specific heat, thermal conductivity, and thermal diffusivity of the plate material is plotted in Fig. 1. As $(Fo)_{1/2}/(Fo)_{\Delta} = \tau_{1/2}/\tau_{\Delta}$, the graphs in Fig. 1 make it possible to determine the thermophysical properties by experimentally measured quantities and, with small values of Bi ($Bi < 0.3$), it is not necessary to find the value of R.

The procedure described for determining the thermophysical properties was applied in the version of the sample being investigated — an anode, heated up with a modulated electron beam. A stepwise change of the thermal flux was accomplished by varying the anode voltage, supplied by a separable vacuum diode. The quantities necessary for calculating the thermal conductivity (see Fig. 1a), thermal diffusivity (see Fig. 1b) and specific heat (see Fig. 1c) of the material of the sample being studied are determined by the experimentally found ratio $\tau_{1/2}/\tau_{\Delta}$.

For metallic samples at a temperature not in excess of 1000°C, the estimated error for this version amounted to 5, 6, and 12% for the specific heat, thermal diffusivity, and thermal conductivity, respectively.

In order to verify the described procedure, the thermophysical properties of tungsten and molybdenum were measured over the range 550-1000°C. The data obtained coincide well with the published data [3] for these materials.

The measurement of the thermophysical properties of polycrystalline cast uranium with 99.75 mass % purity and density 18.8 g/cm³ was carried out on a sample in the form of a disk with diameter 10 mm and thickness 1.2 mm. At temperatures up to 1000°C, the criterion Bi determined from the graphs of Fig. 1 from the ratio found for $\tau_{1/2}/\tau_{\Delta}$, does not exceed 10^{-2} (this means that the calculation of the thermophysical properties of the material are derived independently of R). The measured thermal conductivity and thermal diffusivity, and specific heat of uranium (Fig. 2) coincide quite well with the data of [2], according to which the thermophysical properties of uranium are almost independent of the production technology and preliminary heat-treatment of the material.

The verification carried out provides the basis for assuming that the proposed method can be used for measuring the thermophysical properties of reactor materials.

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PRODUCTION OF INTENSE MONOCHROMATIC BEAMS OF LONGWAVE NEUTRONS FROM THE OPEN TANGENTIAL CHANNEL OF A NUCLEAR REACTOR

B. N. Goshchitskii and V. G. Chudinov   UDC 539.125.5.171

In many physical experiments with neutron beams, it is necessary to use quite intense monochromatic beams of longwave neutrons ($E_N \leq 0.005$ eV). These monochromatic beams of longwave neutrons can be obtained over different ranges of wavelengths by means of polycrystalline scatterers and filters, installed in the open tangential channel of a reactor.

The use of an open tangential reactor channel makes it possible to obtain monochromatic beams of longwave neutrons almost without loss of intensity. A specific geometry of the scatterer is achieved in an open tangential channel in which a scatterer is positioned, and there is no direct "shooting-through" of neutrons from the walls in the outlet section of the channel. Neutrons from the reflector are incident on the scatterer with different directions of motion but the scattered beam is extracted in a given direction which, because of selected collimation, almost coincides with the direction of the channel axis. The initial neutrons incident on the scatterer cannot have a direction of motion along the axis of the channel. This direction appears only during scattering. In fulfilling the relation $t/L_{tr}(E_N) \sim 5$ [t is the scattering length and $L_{tr}(E_N)$ is the transport scattering length of the neutrons], almost all the neutrons of specified energy, incident on the weakly absorbing scatterer, are extracted from the channel.

When using polycrystalline materials as a scatterer, with low absorption elastic noncoherent and inelastic scattering cross sections, and a large elastic coherent scattering cross section, the condition $t/L_{tr} \sim 5$ will not be satisfied simultaneously for neutrons of all energies even at quite large values of t. Neutrons with a wavelength of $\lambda_1 \geq 2d'_{max}$ cannot change the direction of motion in such a way as to move in a direction close to the channel axis. Thus, the energy spectrum of the neutrons at the channel outlet will be chopped from the side of large wavelengths ($\lambda \geq \lambda_1$), while neutrons with $\lambda < \lambda_1$, as indicated, can be extracted almost without loss (Fig. 1a).

If now this beam of neutrons passes through a normal polycrystalline filter located at the outlet from the channel, with $d''_{max} < d'_{max}$, then the neutron spectrum $\Phi(\lambda)$ following it will be chopped from the side of high energies, as neutrons with $\lambda_2 \leq 2d''_{max}$ — for which the elastic noncoherent scattering cross section differs from zero — will be removed from the beam (Fig. 1b). By selecting the scatterer — filter pairs, monochromatic beams of longwave neutrons can be obtained in various ranges of wavelength $\lambda_1 - \lambda_2$. It is obvious that in order to reduce the admixture of neutrons with $\lambda > \lambda_1$ in the monochromatic beam obtained in this way, and which is due mainly to an inelastic scattering process, the scatterer should be cooled. Later, it will be shown that the fraction of admixed neutrons also can be reduced by using scatterers of relatively small length $t \sim L_{tr}$, but in this case the intensity of the useful beam is lost somewhat.

In order to verify and corroborate these reasonings, special measurements were undertaken in the open tangential channel of the IVV-2 reactor. Figure 1c represents the arrangement of the experiment for producing a monochromatic beam of longwave neutrons and for measuring its energy spectrum. As the scatterers, powders of beryllium oxide (dry density ~ 1.1 g/cm^3; length 120 mm and $2d'_{max} = 4.68$ Å) and graphite (density 1.6 g/cm^3; length 120 mm and 25 mm and $2d'_{max} = 6.76$ Å) were used, and at the channel outlet was installed a polycrystalline beryllium filter (length 150 mm and $2d''_{max} = 3.96$ Å). The spectra were measured by the usual procedure of time-of-flight on a base of 417 cm (the experimental conditions and also the characteristics of the reactor and the tangential channel are described in [1, 2]).

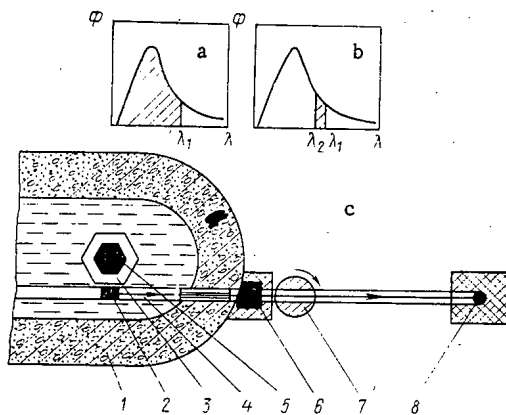Fig. 1   Fig. 2

Fig. 1. Neutron spectra at the outlet from an open tangential channel in which is installed a scatterer (a), and following a polycrystalline filter (b); experimental arrangement (c): 1) reactor shield; 2) scatterer; 3) reflector; 4) tangential channel; 5) active zone; 6) polycrystalline filter; 7) mechanical chopper; 8) detector.
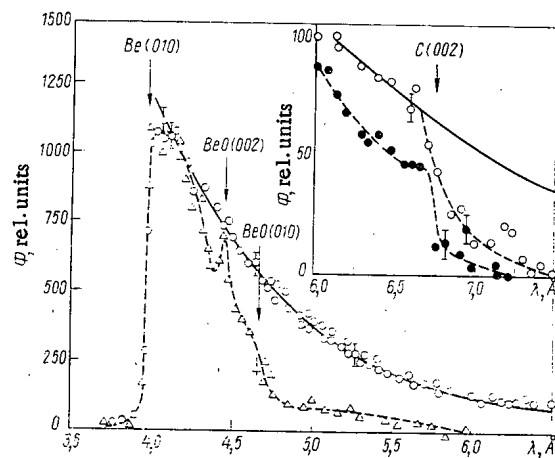
Fig. 2. Spectra of neutrons $\Phi(\lambda)$ extracted from the open tangential channel. Polycrystalline filter — beryllium; scatterer: $\triangle$) beryllium oxide; $\bigcirc$) graphite (t = 120 mm); $\bullet$) graphite (t = 25 mm); ——) Maxwell distribution for $T_0 = 320°K$; – – – –) experiment.

Figure 2 shows the neutron spectra obtained in these measurements. It can be seen that in the long-wave region there is a clearly expressed cutoff when $\lambda = 4.68$ and 6.76 Å for the scatterer of BeO and graphite with a different length, respectively. The equilibrium spectrum of the thermal neutrons in the reactor reflector, calculated for a temperature of 320°K shows the same; this coincides well with the experimentally measured data for a scatterer of graphite (t = 120 mm) over the range of threshold wavelengths from $\lambda_1 = 6.76$ Å to $\lambda_2 = 3.96$ Å. The neutron admixture with $\lambda > \lambda_1$ for beryllium oxide amounts to 10% approximately of the number of these neutrons in the equilibrium spectrum. (It should be noted that the scatterer of graphite with t = 120 mm extracts almost all the thermal neutrons from the channel [2]). The certain difference between the spectra for BeO and equilibrium in the region of 3.96 Å $< \lambda < 4.68$ Å is because in the scatterer of beryllium oxide with t = 120 mm, the condition $t/L_{tr} \sim 5$ which is necessary for the complete extraction of neutrons with the specified wavelength [1] is not fulfilled for all wavelengths.

When the graphite scatterer is installed in the channel, the neutron admixture with $\lambda > \lambda_1$ amounts to ~25% with a length of 120 mm and ~5% with a length of 25 mm (t ~ L_{tr}) of the number of these neutrons in the equilibrium spectrum. At the same time, by reducing the length of the graphite scatterer from 120 to 25 mm, the yield of useful neutrons with $\lambda < \lambda_1$ is reduced by a factor of only 1.5 approximately.

Thus, by using a specific geometry of the neutron scatterer in the open tangential channel, an intense beam of longwave neutrons with a small energy scatter can be obtained immediately at the outlet from the channel, which enables experimental facilities to be constructed with good resolving power and a high transmission.

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SIMPLIFIED DETERMINATION OF THE NUMBER OF
FISSION NEUTRONS EMITTED PER THERMAL NEUTRON
ABSORBED IN A URANIUM — WATER LATTICE

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UDC 621.039.51

One of the basic nuclear physics parameters of thermal reactors is the number of fission neutrons emitted per thermal neutron absorbed:

$$\eta = \frac{\overline{(\nu_f \Sigma_f)_0}^{core}}{(\overline{\Sigma_a})_0^{core}} = \frac{\int_0^{E_{lim}} \int_{V_0} \nu_f \Sigma_{f_0}(E) \Phi_0(r, E)\, dV\, dE}{\sum_i \int_0^{E_{lim}} \int_{V_i} \Sigma_{a_i}(E) \Phi_i(r, E)\, dV\, dE},$$ (1)

where E_{lim} is the limiting energy of the thermal region, V_0 is the fuel volume, and Φ_0 is the neutron flux in the fuel. The summation in the denominator is extended over all the core components. An accurate calculation of the parameter η is a complex and difficult problem.

In many cases, however, it is necessary to find simpler sufficiently accurate methods of calculating η based on reference data obtained by accurate calculations.

One such method is described in the present paper.

For an elementary cell consisting of fuel, moderator, and fuel element cladding (subscripts 0, 1, 2, respectively) Eq. (1) can be reduced to the form

$$\eta = \frac{\overline{(\nu_f \Sigma_f)_0}}{\Sigma_{a_0} + \Sigma_{a_1} \frac{V_1}{V_0} \frac{\overline{\Phi_1}}{\overline{\Phi_0}} + \Sigma_{a_2} \frac{V_2}{V_0} \frac{\overline{\Phi_2}}{\overline{\Phi_0}}}.$$ (2)

The accuracy of the calculation of η depends on the accuracy of the calculation of the thermal neutron fluxes $\overline{\Phi}_i$ averaged over the zones, and the method of averaging the absorption cross sections $\overline{\Sigma}_{ai}$ and the product $(\nu_f \Sigma_f)_0$.

The ratio of the average fluxes calculated by the integral method [1] is in good agreement with experiment. Therefore the accuracy of the calculation of η is determined mainly by the method of averaging the macroscopic cross sections.

In the method of averaging the cross sections which takes account of the effects of heterogeneity [2] the temperature of the neutron gas is determined for each region separately, and the cross sections pertain just to this temperature. It is assumed that the neutrons have a Maxwellian distribution in each zone of the cell.

The following procedure for averaging the cross sections, based on the present method, was used in the calculations. The temperature of the neutron gas in the moderator T_{ng_1} can be determined from the expression

$$T_{ng_1} = T_1 \left[1 + \frac{9.1}{\gamma_1} \vartheta_1 \right],$$ (3)

where T_1 is the temperature of the moderator, γ_1 is the density of water at the temperature T_1 relative to its density at $T = 293.6°K$;
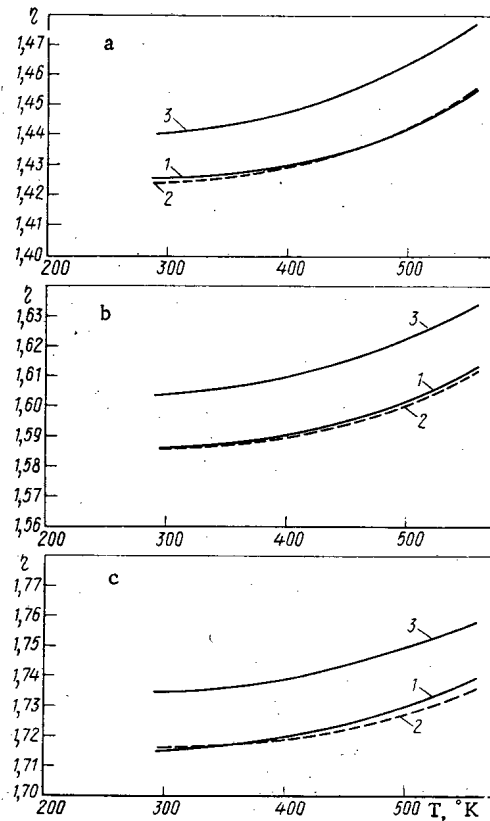

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Fig. 1. Comparison of values of η calculated by using various models for uranium enrichments of a) 1.6; b) 2.4; c) 3.6%.

$$\vartheta_1 = \frac{1}{3.07}\left[\sum a_1 + \sum a_2 \frac{V_2}{V_1} \frac{\overline{\Phi}_2}{\overline{\Phi}_1} + \sum a_0 \frac{V_0}{V_1} \frac{\overline{\Phi}_0}{\overline{\Phi}_1} \right] \tag{4}$$

is the hardness factor of the spectrum; 3.07 is the numerical value of the parameter characterizing the average energy transfer in the collision of a neutron with a moderator nucleus determined by the Nelkin model;

$$\sum a_i = \sum a_i^{2200} \frac{1}{\tilde{v}} ; \tag{5}$$

Σ_{ai}^{2200} is the absorption cross section in the i-th zone, aside from the fuel (UO$_2$), for the tabulated neutron velocity v_T = 2200 m/sec;

$$\sum a_0^{2200} = \sum a_5^{2200} g_a^5 + \sum a_8^{2200} + \sum a_{0_2}^{2200} , \tag{6}$$

where g_a^5 is a factor taking account of the deviation from the 1/v law of the absorption cross section of ^{235}U, taken at the average velocity of the neutron gas in the fuel $\overline{v}_0$ [2] (we assume that ^{235}U is the only absorber in the system which does not follow the 1/v law); $\tilde{v}_i = \overline{v}_i / vT$ is the average velocity of the neutron gas in the i-th zone relative to the tabulated velocity v_T = 2200 m/sec, and $\overline{\Phi}_i$ is the average neutron flux in the i-th zone.

The velocity of the neutron gas in the moderator relative to the velocity v_T = 2200 m/sec is

$$\tilde{\overline{v}}_1 = 1.128 \sqrt{\frac{T_{ng_1}}{293.6}} . \tag{7}$$

The quantity $\tilde{\overline{v}}_0 - \tilde{v}_1$ which takes account of the hardening of the neutron spectrum in the fuel was determined by the equations [2]

$$(\tilde{\overline{v}}_0 - \tilde{v}_1) = 0.3\beta; \tag{8}$$

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$$\beta = \frac{d_0 \, \overline{\Sigma}_{a_0} (1 - P_c)}{\left(1 - \frac{\Sigma_{S_0}}{\Sigma_0} P_c\right)}, \tag{9}$$

where d_0 is the diameter of a fuel slug in cm, $\Sigma_0 = \overline{\Sigma}_{a0} + \Sigma_{S0}$ is the total interaction cross section in cm^{-1} in the "0" zone, and P_c is the probability that a neutron produced in the fuel experiences its first collision in the fuel.

P_c was calculated from the relation [3]

$$P_c = \frac{2d_0 \, \Sigma_0}{2 + d_0 \, \Sigma_0} - \frac{d_0 \, \Sigma_0}{3 + d \, \Sigma_0}. \tag{10}$$

The quantity $\tilde{v}_2$ is defined as the arithmetic mean of the neutron velocities in the fuel $\tilde{v}_0$ and moderator $\tilde{v}_1$.

The method is iterative. In the first approximation we take

$$\frac{\overline{\Phi}_2}{\overline{\Phi}_1} = \frac{\overline{\Phi}_0}{\overline{\Phi}_1} = 1; \quad \tilde{v}_1 = 1.128 \sqrt{\frac{T_1}{293.6}};$$

$$g_a^5 = g_a^5 (kT_1).$$

In subsequent approximations the ratios of the average fluxes are determined by the Amouyal, Benoist, Horowitz method with cross sections calculated by Eq. (5). The calculation is repeated until the difference between the $(m+1)$-th and m-th approximations for $\overline{\Phi}_i / \overline{\Phi}_1$ is small enough. Generally two or three iterations are sufficient.

Our method was compared with others by calculating values of η for 1.6, 2.4, and 3.6% enriched uranium and water temperatures of 293, 373, 423, and 558°K. The optical thickness of a slug $d_0 \Sigma_{a0}^{2200}$ is 0.489, 0.579, and 0.713, respectively.

The fuel was UO_2 with a density of 10.4 g/cm^3. The outside diameter of the fuel element was 9.1 mm, the diameter of the fuel was 7.75 mm, and the cladding was a zirconium alloy. The water — uranium ratio was 1.79. Figure 1 compares the values of η obtained by using various models.

Curve 1 corresponds to the following method of calculating η: thermalization in water is described by the heavy gas model, and the spatial distribution of the neutron flux at each energy is calculated by the method of first collision probabilities. This method was checked against calculations by the THERMOS program [2] for typical lattices with a light water moderator. The results of the calculations are in good agreement.

Curve 2 corresponds to the results of calculations by our method.

For comparison, curve 3 shows values of η calculated in the following way:

The cross sections of all the core components were averaged over a Maxwellian spectrum for a homogenized cell with the same neutron gas temperature over the whole cell [4]:

$$T_{ng} = T_1 \left[1 + C \frac{\overline{\Sigma}_a (kT_1)}{\xi \Sigma_s} \right], \tag{11}$$

where T_1 is the average temperature of the moderator,

$$\overline{\Sigma}_a (kT_1) = \frac{\sum_i V_i \Sigma_{a_i}^{2200} \sqrt{\frac{293.6}{T_1}} \, \overline{\Phi}_i}{\sum_i V_i \overline{\Phi}_i}$$

is the average absorption cross section at the energy kT_1,

$$\xi \overline{\Sigma}_s = \frac{\sum_i V_i (\xi \Sigma_s)_i \, \overline{\Phi}_i}{\sum_i V_i \overline{\Phi}_i}$$

is the average slowing down power at 1 eV, C is a constant depending on the kind of moderator and is equal to 1.88. The flux ratio $\Phi_1/\Phi_0 = 1.1$. A comparison of the results obtained by various methods shows that the points of curve 2 for enrichments of 1.6 and 2.4% and all values of the temperature differ by no more than

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0.097% from curve 1 which corresponds to the most accurate method of calculating η. For an enrichment of 3.6% the maximum divergence of the curves is ~0.23%. It is evident that this stems from the fact that for a slug of such an optical thickness the actual neutron spectrum in the cell (primarily in the fuel) is appreciably different from Maxwellian, and therefore the method of [2] is not quite correct.

The difference between curves 3 and 1 is ~1.5%. Of this difference ~1% is due to the difference in $\overline{\Sigma}_{ai}$ and ~0.5% to the difference in the flux ratios $\overline{\Phi}_i/\overline{\Phi}_0$. Such a large difference is inadmissible.

From a comparison of the values of η obtained by various methods we can recommend the proposed method for performing engineering calculations of uranium — water lattices.

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EXPERIENCE GATHERED IN THE UTILIZATION OF LOW-ENERGY ACCELERATORS FOR THE ACTIVATION ANALYSIS OF METALLURGICAL PRODUCTS

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UDC 543.53

Technological problems encountered in the production of many new materials, particularly refractories and heat-resistant or hard alloys, cannot be successfully solved unless the influence of other elements present either as alloying admixtures or as unwanted impurities upon the new materials is properly assessed.

Certain relationships could be established in various production processes of the Uzbek Combine of Refractories and Heat-Resistant Metals when the above problems were studied with neutron activation analysis using a low-energy accelerator for charged particles as the activator (NG-200 generator of fast particles).

Methods were developed for determining the concentration of O, Si, Y, Fe, and Mo in tungsten and tungsten-base alloys and in several other heat-resistant and high-melting materials. Important characteristics such as performance figures, accuracy, and sensitivity of the methods were studied.

In the application of low-voltage accelerators there arise specific requirements in regard to neutron flux monitoring, sample shape and size, standards, and methods of preparing standards. When an analysis is made

TABLE 1. Comparison of Various Calibration Methods in the Determination of the Yttrium Concentration of Tungsten

Sample No.	Sample wt., g	Y_2O_3 concn., % from $W + Y_2O_3$	Y_2O_3 concn., % from Y_2O_3	Rel. error (%)
1	3,3	2,52	2,08	21,1
2	4,3	2,96	2,4	23,3
3	4,7	2,92	2,34	24,3
4	4,9	2,82	2,24	26,0
5	7,2	3,45	2,6	32,7
6	9,2	2,1	1,4	42,7
7	0,61	2,61	2,02	29,2
8	0,8	2,50	1,94	28,8

TABLE 2. Basic Characteristics of the Activation Analysis of High-Melting Materials ($f \approx 10^{10}$ neutrons/sec)

Material analyzed	Admixtures	Concn. det'd, %	Accuracy of analysis (rel.%)	Sensitivity, %	Time (min) of the determination
Tungsten and tungsten production	O	$10^{-3} - 10^{-1}$	5—15	$(3-5) 10^{-4}$	$\leqslant 5$
	Si	$10^{-3} - 10^{-1}$	10—20	10^{-3}	5—8
	Al	$n \cdot 10^{-2}$	10—20	$5 \cdot 10^{-3}$	25—30
	Mo	$10^{-2} - 1,0$	5—10	10^{-3}	15—20
	Y	$1,0 - 3,0$	3—7	$5 \cdot 10^{-4}$	$\leqslant 5$
	F	$10^{-2} - 1,0$	15—20	$5 \cdot 10^{-3}$	25—30
Molybdenum products	O	$10^{-3} - 1,0$	5—15	$(3-5) 10^{-4}$	$\leqslant 5$
	Fe	$10^{-2} - 10^{-1}$	10—20	10^{-2}	20—30
	W	$10^{-1} - 1,0$	10—20	10^{-1}	5—10
Titanium production	O	$10^{-2} - 1,0$	5—15	10^{-3}	$\leqslant 5$
	Fe	$10^{-2} - 1,0$	10—20	10^{-2}	20—30
	Co	$10^{-2} - 1,0$	10—20	10^{-2}	20—30
	Mo	$n \cdot 10^{-1}$	10—20	10^{-2}	25—30
	Al	$n \cdot 10^{-2}$	25—30	$5 \cdot 10^{-3}$	25—30

on the basis of short-lived isotopes and when pneumatic transportation is used, the system controlling the timing, the sample positioning for irradiation, and the measurements must satisfy increased requirements [1-3].

When materials with large Z are analyzed, one must consider the absorption of both neutrons and γ quanta by interaction with matter. These effects can be eliminated by selecting a corresponding calibration method. In the method employed, synthetic samples in which the material to be examined formed the base material were used as standards.

It follows from the results obtained in determinations of the yttrium concentration in tungsten from the isomer ^{89m}Y ($T_{1/2} = 16.5$ sec and $E_\gamma = 910$ keV) that with pure Y_2O_3 as a standard, the results of the analysis are 20-30% lower than those obtained with standards prepared by introducing yttrium oxide into pure tungsten (Table 1).

Increasing the sample weight means increasing the relative error. The increased errors in the case of samples 7 and 8 are associated with the fact that the weights of those samples were low and that large differences in sample and standard dimensions can influence the results.

A similar method was used to determine the concentration of silicon and aluminum in tungsten-base products (tungsten dioxide, charge and bead). The silicon concentration (in amounts of 10^{-2}-$10^{-4}\%$) was determined from ^{28}Al ($T_{1/2} = 2.3$ min and $E_\gamma = 1.78$ MeV) which is formed by the nuclear reaction $^{28}Si(n, p)^{28}Al$ ($\sigma = 250$ mb); the aluminum concentration was derived from the long-lived isotope ^{24}Na ($T_{1/2} = 15.05$ h and $E_\gamma = 1.37$ and 2.75 MeV) formed by fast neutrons in the reaction $^{27}Al(n, d)^{24}Na$ ($\sigma = 125$ mb) [4, 5].

When a synthetic standard with the material to be investigated as the standard's base is employed, this calibration method is a modification of the addition technique which is widely used in chemical analysis, spectral analysis, and other analyses.

The concentration of fluorine in technological products is of particular importance in tungsten production, because the fluorine strongly influences the properties of the end product and the wear of equipment. Since the checking techniques employed were inadequate, a method based on the use of fast neutrons had to be developed. Since the fluorine is determined via measurements of the positronium activity of ^{18}F ($T_{1/2} = 112$ min and $E_\gamma = 0.511$ MeV), one must consider the possible influence of the radionuclides ^{91}Mo ($T_{1/2} = 15.7$ min), ^{62}Cu ($T_{1/2} = 9.8$ min), ^{64}Cu (12.88 h), and ^{63}Zn (38.4 min), and of several other elements which are positron emitters. But the large differences in the half-lives make it possible to separate the ^{18}F activity in its pure form.

This calibration technique was also used in the determination of the iron concentration of titanium products via ^{56}Mn ($T_{1/2} = 2.56$ h and $E_\gamma = 0.846$ MeV). In this case the monitoring, i.e., the correction for the neutron flux difference between sample and standard, was based upon the ^{48}Sc activity ($T_{1/2} = 1.83$ d and $E_\gamma = 1.314, 1.04, 0.988$ MeV) which arises from the nuclear reaction $^{48}Ti(n, p)^{48}Sc$ ($\sigma = 60$ mb). The sample to be analyzed serves as the monitor of the neutron flux.

The ^{48}Sc half-life differs considerably from the ^{56}Mn half-life, but the results of investigations show that a sufficiently high accuracy can be reached. The authors of [6, 7] could draw similar conclusions regarding the applicability of an analogous method.

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When the γ spectra of irradiated samples are analyzed with a semiconductor detector (specifically in the case of the titanium production), one can determine elements such as Co, Mo, Fe, Al, and others. Table 2 lists the basic features of methods developed for the NG-200 neutron generator (Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR) and tested in the analysis of products supplied by the Uzbek Combine of Refractories and Heat-Resistant Metals.

Let us note in conclusion that at the present time in industry, installations of the above-described type are used only to determine the oxygen concentration of various objects. But in the course of the last few years the combined work of the Institute of Nuclear Physics of the Academy of Sciences of the Uzbek SSR, the Chirchik Branch of the VNIITS, and the Uzbek Combine of Refractories and Heat-Resistant Metals has shown that activation analysis with a neutron generator is promising and that such activation—analytical methods should be introduced into production. One of the main advantages of activation techniques over the conventional chemical and spectral methods employed in Central Laboratories of Factories resides in the speed of activation analysis, because usually short-lived isotopes are used for the analysis.

The availability and the introduction of NG-150I and NG-150M neutron generators with neutron fluxes of the order of $5 \cdot 10^{10}$-10^{11} neutrons/sec and of activation—analysis equipment with which adequately high sensitivity at satisfactory accuracy and high selectivity can be obtained favor an extended implementation of fast-neutron activation analysis in the work of industrial enterprises.

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MEASUREMENT OF THE EFFECTIVE RESONANCE
INTEGRAL OF THORIUM METAL

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We present the results of an experimental determination of the geometric dependence of the effective resonance integral of thorium metal. It was shown in [1] that the dependence of the effective resonance integral on $\sqrt{S/M}$ differs from the linear law $I_{eff} = a + c\sqrt{S/M}$ and that this difference is greatest for small values of the surface to mass ratio S/M. Therefore we measured I_{eff}^{Th} over an appreciably broader range ($0.0778 \leq S/M \leq 0.4610$) than was used in earlier experiments [2, 3].

The effective resonance integral was measured by the activation method using thorium metal rods of five different diameters.

The neutron capture rate in thorium was determined from the intensity of the 311.9 keV γ radiation emitted by ^{233}Pa. The effective resonance integral was determined by comparing the rate of capture of neutrons with energies above the effective cadmium cutoff for a foil irradiated in a thorium rod and a standard thorium foil irradiated in the moderator.

In this case I_{eff}^{Th} is given by the expression

$$I_{eff}^{Th} = \frac{A_{bl}}{A_{st}} I_{\infty}^{Th} K_{bl} K_f \xi, \tag{1}$$

where A_{bl}/A_{st} is the ratio of the intensity of the γ radiation from the foil irradiated in the thorium rod to that from the standard, I_{∞}^{Th} is the true thorium resonance integral, K_{bl} is a normalization factor taking account of the blocking of nuclei in the standard, K_f is the ratio of the neutron flux in the thorium rod where the foil was irradiated to the flux at the standard foil, and ξ is a factor correcting for deviations from the Fermi spectrum.
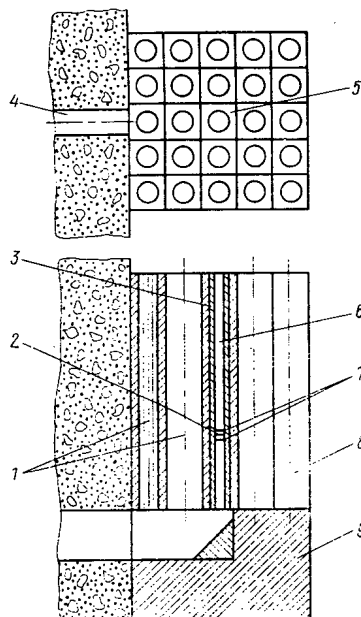

Fig. 1. Schematic diagram of experimental arrangement: 1) RBMK-type fuel assembly; 2) thorium foil; 3) removable graphite inserts; 4) GÉK reactor; 5) channel for irradiating standard; 6) thorium rod; 7) thorium washers; 8) U—C subcritical assembly; 9) converter.

TABLE 1. Result of Measurements

S/M	A_{bl}/A_{st}	ξ	K_{bl}	K_f	I_∞^{Th}, b	I_{eff}^{Th}, b
0,0778	0,1676±0,0022	0,9545				9,68±0,41
0,1024	0,1753±0,0020	0,9667				10,25±0,43
0,1616	0,1914±0,0022	0,9898	0,7246±0,0131	0,9818±0,0039	85±3 [7]	11,46±0,47
0,2905	0,2259±0,0029	0,9921				13,56±0,57
0,4610	0,2608±0,0030	0,9971				15,73±0,65

The foils were irradiated in an experimental arrangement (Fig. 1) on a horizontal channel of the IRT reactor. The arrangement consists of a graphite prism designed to transform a horizontal beam of neutrons into a vertical, and a uranium — graphite subcritical assembly to form a flux of neutrons having nearly a Fermi spectrum [4] at the places where the thorium rod and standard foil are irradiated. The thorium rod under investigation was placed in the central channel of the assembly with a 100-μ-thick thorium foil between two thorium washers. The foils and the washers were completely shielded by 500-μ-thick cadmium filters. The 100-μ-thick standard foil in its cadmium cover and the thorium foil were placed respectively in the moderator and in the thorium rod at the same level in a region with a stabilized neutron spectrum.

The blocking coefficient of the "finite" thickness standard foil was determined from the relation

$$K_{bl} = \frac{R_{Cd}^\infty - 1}{R_{Cd} - 1},\qquad(2)$$

where R_{Cd}^∞ and R_{Cd}^{st} are, respectively, the cadmium ratios of an "infinitely" thin foil and the standard measured at the place where the standard is irradiated. The value of K_{bl} was determined for several "infinitely" thin foils of densities from 0.2 to 0.7 mg/cm^2 produced by electrolytic deposition of $Th(NO_3)_2$ on an aluminum substrate [5]. The measured values of K_{bl} agreed within the limits of experimental error ($\Delta K_{bl}/K_{bl} = 0.0181$).

The correction factor ξ which takes account of deviations from the Fermi spectrum at the locations of the thorium rod and the standard foil was calculated. The neutron spectrum was calculated by the FFK program which solves the Peierls equation in 26 groups [6].

The intensity of the γ radiation from the irradiated foils was measured with a Ge(Li) spectrometer. Since there is a pronounced difference between the spatial distributions of the intensity of the γ radiation from the foils irradiated in the thorium rods and in the moderator, the zonal characteristic of the detector was flattened. The γ spectrum was processed on a computer by approximating the experimental points of the ^{233}Pa photopeak by a Gaussian and interpolating linearly for the background under it by the method of least squares.

The values of the quantities entering Eq. (1) and those obtained for the effective resonance integral are listed in Table 1. The main contribution to the error of the experimentally determined values of I_{eff}^{Th} (> 50%) comes from the uncertainty in the true resonance integral. The errors in the values of I_{eff}^{Th} introduced by the nonuniformity of the zonal characteristic of the Ge(Li) detector and the contribution of the γ radiation of the ^{232}Th fission products in the region of the ^{233}Pa photopeak are insignificant and were not taken into account in estimating the total error.

Two kinds of relations were obtained after processing the experimental values of I_{eff}^{Th} by the method of least squares:

$$I_{eff}^{Th} = (5.41 + 15.16 \sqrt{S/M}) \pm 4.1 \text{ and}$$
$$(4.48 + 16.12 \sqrt{S/M + 0.026}) \pm 4.1\%.$$

The parameters in these relations are in good agreement with those calculated in [1]:

$$(I_{eff}^{Th} = 4.96 + 15.46 \sqrt{S/M} \text{ and}$$
$$4.56 + 15.68 \sqrt{S/M + 0.020}).$$

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CALCULATION OF β-PARTICLE TOTAL BACKSCATTERING COEFFICIENT FOR THICK ABSORBERS

V. A. Kuz'minykh and S. A. Vorob'ev   UDC 539.121.72

This paper gives values of the backscattering coefficient obtained from calculation of β-particle transport through matter by means of the segment model [1, 2]. The calculation includes fluctuation in energy losses; multiple scattering of the particles was computed by the method of Goudsmit and Saunders using the Spenser differential cross section [3]. The method for taking the energy spectrum of the source into account is described in [4]. The limiting energy at which the calculation was cut off was 10 keV.

The calculation was performed for radioisotopic sources in the range of maximum β-particle energies from 309 to 1707 keV for electrons and from 542 to 1894 keV for positrons. The energy spectrum of the sources was calculated from the theory of β decay with allowance for the electric field of the atom [5]. The angular distribution of the β-particle flux was assumed isotropic in a 2π solid angle. Calculated results for the total backscattering coefficients for electrons and positions are given in Table 1 for thin sources on semi-infinite backings.

As the maximum energy of the β particles from a radioisotope increases, the backscattering coefficient for electrons increases somewhat but the dependence on E_{max} is weak; the backscattering coefficient for positrons is practically independent of β-particle maximum energy. In the case of normal incidence of beam particles, the energy dependence of the backscattering coefficient is more clearly expressed and is seen as a linear decrease of the backscattering coefficient as E_{max} increases. Apparently, the energy dependence $\eta(E)$ for an isotropic source of β particles is compensated by the dependence of the total backscattering coefficient on the angle of incidence θ of the particles at the absorber, which depends on energy in a complex manner. It was shown [6] that the backscattering coefficient increases more rapidly with increasing θ at high particle energies.

Figure 1 shows computed curves of $\eta(Z)$ for (a) ^{170}Tm electron and (b) ^{69}Ge positron sources. The values obtained for the backscattering coefficient fall above the experimental points. If one neglects the weak dependence $\eta(E_{max})$, the results presented for isotropic β sources can be described by the following expressions:

$$\eta^- = 0.2290 \lg Z + (0.1045 \pm 0.02);$$
$$\eta^+ = 0.2395 \lg Z + (0.0625 \pm 0.02). \tag{1}$$

TABLE 1. Values of Total Backscattering Coefficients for Electrons and Positrons, %

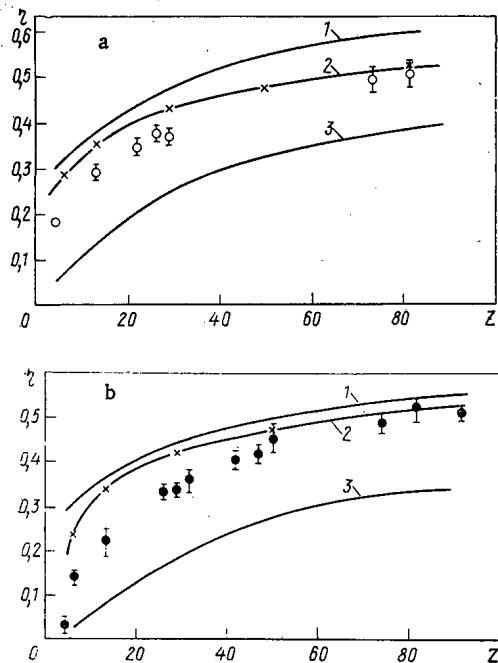| Radionuclide	E_{max}, MeV	Absorber material				
		C	Al	Cu	Sn	Pb
^{60}Co	0,306	29,57	35,75	42,8	47,51	—
^{204}Tl	0,765	28,75	35,33	43,05	48,2	52,15
^{170}Tm	0,950	28,6	35,21	43,2	49,7	53,0
^{137}Cs	1,180	29,6	35,74	43,3	48,4	52,05
^{89}Sr	1,462	—	35,03	43,8	49,75	54,9
^{32}Si → ^{32}P → ^{32}S	1,707	27,65	35,0	44,0	50,3	55,8
^{22}Na	0,540	24,8	34,2	42,15	47,2	51,0
^{64}Cu	0,656	23,8	33,9	42,4	48,0	51,55
^{69}Ge	1,213	—	34,1	42,5	28,05	52,25
^{68}Ga	1,894	23,0	33,35	42,4	47,7	52,2

Fig. 1. Calculated data for backscattering coefficient: 1) isotropic source with an energy equal to the mean energy of the spectrum; 2, 3) isotropic and normal incidence on absorber of particles with a continuous spectrum; ○, ●) experimental points [7, 8] for electrons and positrons, respectively.

For a thin radioisotopic source on a massive backing, the particle flux intensity $I_{2\pi}$ into a hemisphere depends on the backing material. Using the empirical relations (1), one can write expressions for the activity of a β source:

$$A^- = I_{2\pi} (0.1145 \lg Z + 0.5522 \pm 0.01)^{-1};$$
$$A^+ = I_{2\pi} (0.1197 \lg Z + 0.5312 \pm 0.01)^{-1}. \tag{2}$$

The curves 1 in Fig. 1 were obtained for the average energies of the corresponding nuclides. In this case, the particle flux is isotropically incident on the absorber in a 2π solid angle. The curves 3 were obtained for monodirectional spectral sources. The dependence of the total backscattering coefficient on Z is smooth and the calculated curves are mutually parallel for the geometries given. For monodirectional beams, the backscattering coefficient is considerably lower than for isotropic beams since in the latter case the backscattering coefficient for particles incident on the absorber at large angles θ makes a significant contribution to the total backscattering coefficient.

Values of the backscattering coefficient therefore depend on the experimental geometry. Furthermore, the differences in the interaction of electrons and positrons with matter are expressed differently in backscattering. The ratio η^-/η^+ has its highest value for normal incidence of particles on an absorber and the difference in the backscattering coefficients for electrons and positrons is insignificant for isotropic incidence.

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RANGE OF FAST ELECTRONS IN DIELECTRICS

O. B. Evdokimov, B. A. Kononov, and N. I. Yagushkin

Irradiation of dielectrics by fast electrons and β particles is accompanied by considerable buildup of space charge before electrical breakdown [1, 2]. The electric field which is created markedly reduces the thickness of the layer in which the energy of the electrons is absorbed; this is important in evaluation of the radiation effect of charged particles on high-resistance materials. A reduction of the electron range in organic polymers was measured [3, 4]. A theoretical estimate of the kinetics of electron range during irradiation was made [5], but it does not include relaxation of space charge during irradiation. Under the same approximation, an expression was obtained [6] for the range of heavy nonrelativistic charged particles.

This paper presents a calculation of the kinetics of the range of fast electrons in a slab of dielectric, the thickness of which is several times greater than the value of the range (d ≫R), as a function of irradiation time with induced conductivity taken into account.

We assume that on the external surface (x = 0) of a dielectric (x ≥ 0) there is normally incident a beam of electrons with an energy T_0 and a current density I_0 ($I_0 < 0$). Thermalizing in the material, the electrons are captured by traps and create an electric field E (x, t) where t is the irradiation time. According to estimates, the time for thermalization and capture of electrons in a dense medium is less than 10^{-9} sec. If the beam is not confined by its own space charge, the typical time for variation of the field in this case is greater than the thermalization time and one can therefore assume the electron moves in a static field (quasistationary approximation). We further consider the range of electron energies (hundreds of keV and above) where the mean specific energy loss B(T) is weakly dependent on the energy T:

$$T_0 \approx <B> R(t) + e \int_0^{R(t)} E(x, t)\, dx \equiv <B> R(t) + eU(t), \tag{1}$$

where $\langle B \rangle = T_0 R^{-1}(t=0)$; R(t) is the range at the time t; e > 0.

Equation (1) refers to those electrons which are slightly scattered and therefore reach depths corresponding to the range. The dynamics of the field E (x, t) are determined by the equation for the total current J(t):

$$\frac{\partial \varepsilon E(x, t)}{\partial t} + j_c(x, t) + j_0(x, t) = J(t), \tag{2}$$

where j_0(x, t) is the beam current density at the depth x; j_c(x, t) is the conductivity current; J(t) ≈ 0 since d ≫ R; ε is the dielectric constant (it is assumed ε = const). It is known from studies of radiation conductivity [7, 8] that the specific electrical conductivity λ of high-resistance dielectrics is given by the expression

$$\lambda = aD^\Delta, \tag{3}$$

where a and Δ are numbers typical of a given material (0.5 < Δ < 1); D is the energy absorption rate per unit volume of material. Under conditions of space-charge buildup, the distribution D(x) depends on time since the electron flux is exposed to the effects of the field [9]. Then the conductivity current is

$$j_c(x, t) = aD^\Delta(x, t) E = a|\varepsilon^{-1} I_0|^\Delta D_1^\Delta(x, t) E(x, t), \tag{4}$$

where D(x, t) is the distribution of the absorbed energy for a single electron. Equation (4) assumes that the typical time for the establishment of conductivity is much less than the typical time for space-charge buildup.

It is known from work on radiation conductivity that in many cases one can assume the time for establishment of conductivity to be practically instantaneous. The beam current at the depth x is determined by the penetration coefficient h(x, t), which also depends on the field and therefore on the time [9]:
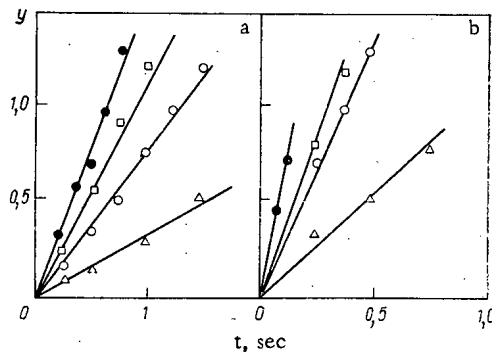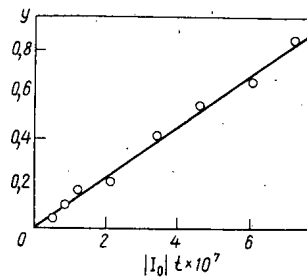

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Fig. 1

Fig. 2

Fig. 1. Dependence of the quantity y on irradiation time plotted from the data of [4] for (a) polymethylmethacrylate and (b) polystyrene and an initial electron energy of 1.3 MeV. Experimental data for $I_0 = 0.5$ ($\triangle$), 1.0 (◯), 1.5 (☐), and 2.0 (●), $\mu A/cm^2$.

Fig. 2. Dependence of y on the introduced charge $|I_0|t$ plotted from the data of [10] for Lucite where $T_0 = 1.5$ MeV and $I_0 = 0.1$ A/cm^2.

$$j_0(x, t) = I_0 h(x, t). \qquad (5)$$

Substituting Eqs. (4) and (5) into Eq. (2), integrating the results with respect to x from 0 to R(t), expressing U(t) through R(t) from Eq. (1), and considering that $dU = -e^{-1} \langle B \rangle \, dR$, we obtain an equation for the range:

$$[e^{-1} \langle B \rangle + E(R, t)] \frac{dR}{dt} +$$

$$+ \left[\frac{a \langle B \rangle |I_0|^\Delta}{\varepsilon e^{1+\Delta}} \langle D_1^\Delta(t) \rangle - \frac{I_0}{\varepsilon} \langle h(t) \rangle \right] \times$$

$$\times R + ae^{-(1+\Delta)} \varepsilon^{-1} T_0 |I_0|^\Delta \langle D_1^\Delta(t) \rangle = 0, \qquad (6)$$

where the angle brackets indicate averaging over x in the range $0 \le x \le R$. Averaging of h is performed with a weight of 1 and of D_1^Δ with a weight $E(x, t)$.

Determination of the field $E(R, t)$ is a complex problem. For light materials, however, it is known that the main portion of the electrons is thermalized near $x \sim R$. This effect is evidenced even more strongly in the presence of a retarding field [9]. For this reason, and because $D_1^\Delta(x)$ is a slowly varying function of x [9], one can assume $E(R, t) \approx U(t)/R(t)$. The approximation of a uniform field for $x \le R$ was also used in [5]. Using Eq. (1), one can write Eq. (6) in the form

$$dR/R \{R/R_0 + R/r - 1\}^{-1} = -dt/\langle \tau \rangle, \qquad (7)$$

where

$$\frac{1}{r} \equiv \frac{e^{1+\Delta} |I_0^{1-\Delta} \langle h(R) \rangle}{a T_0 \langle D_1^\Delta(R) \rangle},$$

$$\frac{1}{\langle \tau \rangle} \equiv \frac{a |I_0|^\Delta}{\varepsilon e^\Delta} \langle D_1^\Delta(t) \rangle. \qquad (8)$$

Since the field changes insignificantly when $x < R$, the quantities $\langle h \rangle$ and $\langle D_1^\Delta \rangle$ are practically independent of the field distribution and therefore are functions of R. In turn, the dependence of $\langle h \rangle$ and $\langle D_1^\Delta \rangle$ on R is very weak. It then follows from [9] that when $Z = 6$ and $T_0 = 1$ MeV, the increase in the quantities $\langle D_1^\Delta \rangle$ and $\langle h \rangle$ does not exceed 10% for a change in eE from 0 to $0.8 \, MeV/g \cdot cm^{-2}$ so that $\langle \tau \rangle$ and especially r in Eqs. (7) and (8) can be considered independent of R and t. After integration of Eq. (7), we obtain the dependence of the range R(t) on irradiation time:

$$\frac{1}{R} \approx \frac{1}{R_0} + \frac{1}{r} \left[1 - \exp \left(-\frac{t}{\langle \tau \rangle} \right) \right]. \qquad (9)$$

If necessary, the weak dependence of r and $\langle \tau \rangle$ on R can be taken into consideration.

A comparison of calculated results from Eq. (9) with known experimental data [3, 4] obtained for organic polymers is shown in Figs. 1 and 2. The function (9) can be linearized in the form

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$$y(t) \equiv \ln \frac{R(R_0 - R_\infty)}{R_0(R - R_\infty)} = \frac{t}{<\tau>}, \tag{10}$$

where $R_\infty = [R_0^{-1} + r^{-1}]^{-1}$ is the maximum value of the range. Figure 2 shows data from [3] which was analyzed by Kennedy and Kohlberg [10].

Thus Eq. (9) is confirmed by experiments on organic polymers for beam currents from 0.1 to 10^{-7} A/cm^2. One can show that the equation obtained in [5] without consideration of conductivity is a particular case of Eq. (9). In order to predict the thickness of the irradiated layer in a given material, it is necessary to know the parameters a and Δ, which are determined from experiments on radiation conductivity.

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COMECON CHRONICLES

THIRTIETH MEETING OF COMECON STANDING COMMITTEE
ON THE PEACEFUL USES OF ATOMIC ENERGY

Yu. I. Chikul

The COMECON Standing Committee on the Peaceful Uses of Atomic Energy (SCPUAE) held its 30th meeting in Vraca, People's Republic of Bulgaria, in June 1976. Representatives of the international commercial associations Interatominstrument and Interatoménergo took part in the work of the Committee.

The Committee discussed measures in the domain of nuclear power engineering following from the decisions of the COMECON Executive Committee and the COMECON Committees on cooperation in planned activities and on scientific and technical cooperation; it considered preliminary proposals for the atomic energy section of a draft long-term program of cooperation on providing for the economically justified needs of the COMECON member-countries in regard to the principal forms of energy, fuels, and raw materials for the period up to 1990; the Committee also considered programs of scientific and technical cooperation on the development and promotion of computer-assisted systems of inspection and control for an atomic power plant with a VVÉR-1000 (water-moderated, water-cooled) power reactor as well as the development and promotion of methods and instruments for γ-ray resonance (Mössbauer) spectroscopy for the 1976-1980 period, and proposals concerning the forms of cooperation in these areas.

The Committee approved draft "Regulations Concerning the Safe Transportation of Spent Nuclear Fuel from Atomic Power Plants of COMECON Member-Countries. Part One — Transportation by Rail" and decided to present it to the COMECON Executive Committee for approval; these were documents concerning norms and methods of radiation sterilization of medical materials and supplies and recommendations concerning methods for technological dosimetry of radiation installations with radioisotopic γ sources.

The Committee considered a report on the activity of the Bulgarian Committee on Scientific and Technical Cooperation in the 1972-1975 period, discussed various aspects of the further development of cooperation with the International Atomic Energy Agency in connection with the conclusion of an Agreement between COMECON and the IAEA. The meeting approved a report on the work done by the SCPUAE in 1975 and its further activities, and considered questions emerging from the meetings of the working organs of the Committee in the first half of 1976. The Committee adopted suitable recommendations and decisions on all questions considered.

The Committee meeting took place in an atmosphere of friendship, complete mutual understanding, and business-like cooperation.

In the period preceding the Committee meeting, a protocol was signed in Sofia on extending to 1980 the validity of the Agreement establishing a provisional international scientific research team to carry out research on the reactor physics of the critical assembly of the VVÉR. This ensures that important problems concerning the design of a high-power reactor of this type will be resolved by the joint efforts of the COMECON member-countries.

The representatives of the contracting parties of the agreement noted with satisfaction that the National Committee of the Republic of Cuba on the Uses of Atomic Energy had acceded to the agreement in 1976.

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CONFERENCES, MEETINGS, AND SEMINARS

THIRD ALL-UNION SCIENTIFIC AND PRACTICAL
CONFERENCE ON RADIATION SAFETY

V. I. Ivanov and U. Ya. Margulis

The conference, meeting in Moscow in May 1976, was organized by the State Committee of the Council of Ministers of the USSR for Science and Technology and the All-Union Central Council of Professional Societies. It was attended by some 500 persons representing 140 scientific and practical organizations.

Opening the conference, the Deputy Minister of Health Protection of the USSR, A. I. Burnazyan, noted that complex scientific, technical, hygienic, and clinical problems of radiation protection are being successfully solved thanks to the efforts of physicists and engineering, technical, and medical workers. As a result, nucleonics has become a branch of industry that poses minimal danger to man and the environment.

At four sessions the conference discussed 117 papers. More than 100 were presented by rapporteurs in the form of surveys of particular topics.*

The first session heard papers read by the leaders of the various lines of work and programs. A paper by A. P. Semenov and V. I. Ivanov dealt with the results of the fulfilment of a coordinated five-year plan of work on radiation safety. They noted that as a result of the implementation of the plan, significant progress had been made in elaborating the scientific principles and in improving the methods and measures of radiation safety. The elaboration of normative, methodical, and legislative documents was an appreciable contribution to ensuring safe conditions for work with sources of ionizing radiation. Thirty documents were ratified and 12 others were presented for ratification.

Present-day approaches to the convention of standardizing the radiation factor as well as to the principal unresolved problems in this area were presented in a paper by L. A. Il'in and Yu. I. Moskalev.

Realization of the established norms and the requirements with regard to ensuring radiation safety involves a necessarily single approach to the interpretation of these conditions as applied to the system of radiation monitoring. Methodical and technical means of solving this problem were considered in a paper by A. D. Turkin et al.

B. M. Isaev analyzed work aimed at unifying measurements in the domain of radiation safety, elaborating and promoting standard measuring equipment, and organizing the inspection of dosimetric equipment.

A paper by E. E. Kulish discussed the use of radioactive isotopes in the national economy, as well as improvement of the technology of preparing radiation sources so as to increase safety in their extensive application in many spheres of industrial activity.

In their paper O. G. Pol'skii and V. Ya. Golikov dealt with the establishment of a state sanitary inspectorate in the USSR to look after radiation hygiene and the prospects for improving it. Some current aspects of radiation safety were taken up in a paper by P. V. Ramazaev and S. I. Tarasov.

Subsequent sessions considered specific problems of radiation safety.

Serious attention at the conference was paid to ensuring radiation safety in atomic power plants. The papers read analyzed the principal sources constituting the radiation facility in an atomic power plant and their relative contribution to the dose burden received by the personnel. It was shown that the operation of an atomic power plant presents no danger to the personnel and the environment. Thus, the radiation doses received by personnel are, on average, 1-2 rem/yr, and do not exceed 2.5 rem/yr, i.e., are below the regulatory

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*The surveys "Radiation safety at nuclear power stations" and "Radiation safety during operation of high-intensity radiation equipment" are published in this issue on pp. 890 and 897.

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maximum permissible levels. The average ejections of radioactive material are significantly lower than the maximum permissible and do not lead to contamination of the environment.

Since tritium is one of the "critical" isotopes determining the radiation conditions in the region of the site of a nuclear power plant, particular attention was devoted to evaluation of the equilibrium between the tritium concentration and the amount ejected into the environment. Using the example of an atomic power plant with a VVÉR-440 reactor, it was noted that in all cases the tritium content in the air and water does not exceed the mean permissible concentration. The conference also considered the principles and methods of dosimetric monitoring of the tritium content in the premises of atomic power plants, objects in the external environment, and the human body.

Interesting information was given in a paper on the relative hazard which excursions from atomic and thermal power plants constituted for the population of adjoining regions.

Questions of ensuring radiation safety during the use of high-power radiation techniques, gamma flaw detection or defectoscopy, neutron and tritium sources, etc. were discussed in considerable detail. The results were presented of investigations on radiation hazard factors arising during irradiation processes in high-power apparatus as well as in other aspects of the use of isotopic sources. It was noted that as far as ensuring safe working conditions was concerned, industrial radiation -- chemical technology had the advantage over the traditional technology. Individual doses of radiation received by the operating personnel of high-power radioisotope apparatus does not exceed 0.3 of the maximum permissible doses, i.e. 1.5 rem/yr. Papers read at the conference presented material on the optimization of the protective devices of such apparatus. A mathematical model that was proposed enables the probability of a breakdown to be predicted on the basis of data concerning the failure of individual units. The role was established of radiolysis products (ozone, nitric oxides) entering the air of working premises during normal operation of high-power radioisotope apparatus and in breakdown situations. Many years of observation, the results of which were presented at the conference, show that the design of high-power radioisotope apparatus and shielding systems prevent radioactive materials from getting into the surroundings and the personnel from being over-irradiated under normal operation. The optimal level of radiation monitoring was established on the basis of these studies.

At present more than 3000 gamma flaw detectors are in use in the USSR. Therefore, data on experience in ensuring radiation safety during gamma flaw detection were of particular interest to the conference participants. Over the past 10 years the individual radiation doses received by defectoscopists decreased more than six times and amounted to 0.7 rem/yr, going down to 1.5-2 rem/yr in individual cases.

Some papers were devoted to radiation monitoring and its optimal level when using ionizing radiation. Papers on the achievements in the metrology of ionizing radiation presented data on checking meters used to measure the equivalent dose of fast neutrons and gave the characteristic of State standards of equivalent dose of neutron radiation, units of x-ray dose absorbed (20-60 keV), units of brehmsstrahlung energy flux (5-50 MeV), and a special standard of radioactive aerosols.

Considerable attention at the conference was paid to the dosimetry of internal irradiation. Questions of the dosimetry and evaluation of individual doses of external irradiation have, in principle, been resolved. However, difficulties are encountered in estimating the individual doses of internal irradiation owing to the imperfection of methods available for direct determination of the entry of radioactive materials into the body as well as methods of studying the characteristics of aerosol systems and estimating the doses of radiation received by critical organs, etc., on the basis of these characteristics.

Papers pointed to the achievements in ascertaining the parameters of various aerosol systems, on the basis of which it is possible to develop express analyses of internal irradiation. A noteworthy evaluation was made of the role of the statistical laws governing the ingestion of radioactive materials from the air into the human organism upon inhalation of aerosols, as well as during their expulsion from the organism.

Of interest are new data about methods of reliably estimating irradiation doses received by the lungs when radioactive materials lodge in them, methods of radiographic analysis of aerosol samples, and methods based on multicascade impactors. The characteristics were presented of new fibrous filters based on FP (absorption filter) type cloth for analyzing and trapping radioactive aerosols, iodine, and mercury in an aerial environment. Some papers reported on the evaluation of the reliability of measurements of radioactive aerosol concentration, on the physicochemical properties of aerosols, and also on the radiation burden when radioactive materials are inhaled. The conference heard information about new apparatus for monitoring the entry of radioactive materials and new methods of evaluating the content of radioactive materials in biosubstructures.

Significant progress has been made in developing methods for mathematically modeling radionuclide metabolism on the basis of microchamber models.

Extremely useful and practically important information was given by papers on improvements in methods of spectrometry of human irradiation which permit direct determination of the content of radioactive materials in the human body. There have been noteworthy developments in multiwire proportional counters for measuring the ^{239}Pu content in human lungs, spectrometry for determining the total content of γ-emitting isotopes in the human body by longitudinal scanning with four NaI(Tl) detectors, as well as methods for directly determining the decay products of radon in the respiratory organs from the intensity of the γ radiation emitted.

The conference paid particular attention to the principles for standardizing radiation factors. From the present concept of the thresholdless effect of ionizing radiation it follows that even with a small dose of radiation there is a definite probability of adverse results with a stochastic character. In this connection, the concept of risk occupied a significant place in the discussion.

Some papers contained interesting information about the approach to establishing standards for the joint action of radiative and nonradiative time factors, as well as about the health of personnel working with radiation sources for a long time.

The conference noted the high level and scale of work in the realm of radiation safety, bringing an improvement in the conditions of work with radioactive materials and sources of ionizing radiation. It was also noted that the experience of atomic science and engineering leads to the introduction of ionizing radiation sources with new parameters, and this in turn requires new scientific development and research. Attention was drawn to the need for economic optimization of measures to ensure radiation safety.

The conference adopted recommendations aimed at further improving means and methods of ensuring radiation safety.

The conference proceedings will be published in the journal "Izotopy U SSSR" (Isotopes in the USSR).

ALL-UNION CONFERENCE ON THE USE OF
NEUTRONS IN MEDICINE

B. A. Bedrov and V. N. Ivanov

An All-Union conference on "The Use of Neutrons in Medicine," organized by the Scientific-Research Institute for Medical Radiology, Academy of Medical Sciences of the USSR (SRIMR AMS USSR), was held in Obninsk on May 18-19, 1976. Its program comprised five main sections: the physicotechnical aspects of the application of neutrons in radiology and medicine, the biological effect of neutrons, the use of neutrons in clinical practice, the use of methods of neutron activation analysis in the clinic, and ensuring the radiation safety of the operating personnel and patients.

Up to now, questions of the use of heavy nuclear particles, including neutrons, for beam therapy have been in the discussion stage. There are three variants of the application of neutron radiation in oncology: fast-neutron contact radiotherapy and teletherapy, and neutron-capture therapy with intermediate and thermal neutrons. The latter is a combination of irradiation with external beams of neutrons and the introduction of special nuclides into the tumor for an intense local attack on the tumor cells.

The best-developed research has been that on contact neutron radiotherapy. The leading oncological centers in the USSR have begun a clinical study of the transplutonium source ^{252}Cf. A paper by A. G. Sul'kin (All-Union Scientific-Research Institute of Radiation Therapy, ASRIRT) took up questions of technical equipment for contact therapy. Small, manually inserted, pin-type sources (diameter 1.2 mm, active length 10-30 mm) have been constructed. However, most promise is held out by apparatuses with remote insertion

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which are radiation-safest. At the present time a neutron therapeutic apparatus ANET-1 has been developed for contact beam therapy; the experience gained from the operation of this apparatus in the Central Institute of Advanced Training for Doctors (CIATD) was generalized in a paper by A. V. Kozlova et al. The apparatus is expected to be approved early in 1979 for intratumor irradiation with sources of higher activity. To enhance the capabilities of contact therapy, work is under way on flexible sources.

The physical characteristics of ^{252}Cf sources were the subject of papers by S. N. Kraitor et al. (Biophysics Institute of the Ministry of Health of the USSR) and G. P. Elisyutin and V. Ya. Komar (ASRIRT). The results of experimental studies on the microdosimetric characteristics of these sources were discussed by a paper delivered by V. G. Videnskii and V. V. Farnakeev (SRIMR).

An important place in the discussion was taken by questions of dosimetry and radiation monitoring in contact therapy with a ^{252}Cf source. V. N. Ivanov, representing SRIMR, ASRIRT, and the Biophysics Institute of the Ministry of Health of the USSR, reported on the development of a system of dosimetric and radiation monitoring, establishing suitable conditions for irradiating patients according to a prescribed plan, for analysis of clinical and radiobiological results, as well as personnel monitoring. At present, this system is used in applicator and intratissue therapy of patients with tumors of the head and neck. The radiation-hygienic basis for the application of ^{252}Cf in beam therapy was also considered in a paper by A. V. Kozlova et al. The researchers have used ^{252}Cf for applicator neutron therapy of patients with skin cancer and malignant melanoma.

Extremely interesting data about radiation monitoring and protection in intracavitary therapy of oncological--gynecological patients were presented by V. D. Abdullaev et al. from the Kiev Roentgeno-Radiological and Oncological Scientific Research Institute (KRROSRI). Their research made it possible to draw up norms for the work of personnel with ^{252}Cf sources, norms which satisfy indispensable requirements. Thus, a manipulation nurse can carry out no more than six loadings and unloadings a week, and a doctor, no more than ten. Dosimetric prerequisites for intracavitary neutron therapy of uterine cancer by the simple overloading principle were also discussed in a paper by K. N. Kostrominaya et al. (CIATD). On the basis of the experience gained in employing ^{60}Co γ-ray sources, the authors studied different geometries for neutron radiation sources.

Great interest was aroused by papers on the first clinical results of the use of ^{252}Cf (B. M. Vtyurin et al., SRIMR; A. V. Kozlova et al., CIATD; E. S. Kiselev et al., Moscow Oncological Scientific-Research Institute (MOSRI); and V. D. Abdullaev et al., KRROSRI). Contact therapy was administered to 40 patients in SRIMR, 13 in CIATD, 12 in MOSRI, and 12 in KRROSRI. The application of ^{252}Cf for the treatment of radioresistant tumors was found to yield favorable direct results. It was shown that to accelerate the evaluation of the effectiveness of neutron therapy and to determine its place among other methods of beam therapy research equipment must be improved substantially by increasing the number and assortment of sources.

In appraising neutron teletherapy, the papers dealt primarily with the justification of the medicotechnical need to produce beams of fast neutrons in special-purpose installations and experimental physical apparatus adapted for clinical purposes (E. A. Zherbin et al., SRIMR; A. I. Ruderman and G. V. Makarova, Oncological Scientific Center). In her paper, the Oncological Center representative gave data forming the basis of a project for producing a beam by using the fast pulsed reactor in Dubna. Work has begun in KRROSRI on producing for therapeutic purposes a beam of fast neutrons in the U-120 cyclotron of the Institute of Nuclear Research, Academy of Sciences of the USSR (V. N. Letov et al.).

The physicotechnical aspects of neutron-capture therapy employing proton accelerators was considered in a paper by E. A. Zherbin et al. Unfortunately, there were no papers on the chemical problems involved in this form of therapy.

Many papers were devoted to the biological effect of neutrons.

The Scientific Council on Roentgenography and Radiology at the Presidium of the Academy of Medical Sciences of the USSR was authorized to organize work on establishing a scientific—methodological center for the coordination of scientific research on the use of neutrons in biology and medicine and the introduction of the results of scientific work into practice.

THIRD MEETING OF THE IATE TECHNICAL COMMITTEE
ON HIGH-ACTIVITY AND α-EMITTING WASTE

Yu. P. Martinov

The Technical Committee meeting in Vienna, Austria, from May 10-14, 1976 was attended by experts from Britain, India, the USSR, the USA, France, the Federal Republic of Germany, Japan, and some international organizations. The meeting discussed a wide range of topics: national programs, the results of the May, 1976 symposium on the handling of radioactive waste from the fuel cycle, and methods of handling α-emitting waste, means of solidifying waste, and the indices of the relative safety of the waste.

Cooling spent fuel for three years prior to its regeneration and subsequent solidification of the waste was recognized to be optimal for fuel elements of thermal reactors. French and British specialists believe that a shorter cooling time would be preferable for fast-reactor fuel, whereas in Federal Germany it is suggested that the same period be retained in this case. There is no consensus as to the necessity of separating transuranium elements from high-activity waste. Many believe that this does not lead to a reduction in the toxicity of the waste. The question may be resolved after additional information is obtained about the technology of separation in secondary waste, about the hazards and risk in the processes of separation and burial, about the uses of transuranium elements, transmutation, etc.

Many countries are continuing to develop methods of processing and burying high-activity and α-emitting waste. It is considered to be of cardinal importance to ascertain their properties. Methods of reducing bulk by incineration and leaching are being worked out for α-emitting waste containing plutonium. The possibility is being investigated of applying cryogenics for plastic and resins. Experiments are being carried out to ascertain the properties of fuel-element cans, possibilities of deactivating them, reducing the volume occupied, and obtaining alloys at temperatures which are not high, comparatively speaking.

Most success has been attained in solidifying high-activity waste. Processes of calcination, vitrification, and incorporation into a metallic matrix are being worked out. Vitrification is at present regarded by all as the most suitable process for solidification. A plant for vitrifying waste is being developed in a number of countries (in particular, an electromelt furnace in the USSR in which a high output has been attained with a model solution). France, which plans to conduct active operations in 1977, has come closest to carrying out vitrification on an industrial scale.

Under laboratory conditions, new methods are being used to solidify waste: sintering a pressed mixture of calcinate and glass-forming additives, and production of "multibarrier" glasses. Work is under way to improve the properties of the solidified waste by employing directional crystallization.

Geological formations are considered to be the most suitable for burial of solidified high-activity and α-emitting waste. Thus, the United States, e.g., is increasing appropriations for developing methods of final burial, the increase being from 7-34 million dollars. Of this sum, 95% will be used to check out the possibility of burying waste in a geological formation on dry land, 3% on burial in the seas and oceans, and 2% on transmutation and ejection into outer space.

An IAEA official, G. Grover, told of plans to establish a committee on the removal, storage, and burial of gaseous radioactive elements from aerial discharges. The Committee recommended that instead of engaging in work on the development of an index of relative safety, which does not take account of some important properties of waste and burial conditions, the IAEA should concentrate on elaborating a model of the distribution of radionuclides from burial sites of radioactive waste.

Since the extremely broad subject-matter of the May symposium did not permit its participants to discuss some interesting topics, on a motion by the USSR the Committee recommended that a symposium on the handling of high-activity waste be held in 1980.

5TH INTERNATIONAL SYMPOSIUM ON THE DESALINATION OF SEA AND SALT WATER

O. I. Martynova

The symposium was held in Alghero, Sardinia, on May 16-20, 1976, with 530 delegates from 32 countries participating. In all, 114 papers were read and 48 others appeared in abstracts. The Soviet delegation presented 14 papers.

Desalination of sea and salt water in most countries is linked with nuclear power, first because of the construction of atomic power plants which, in addition to generating electricity, will also desalinate water. In existing projects employing thermal methods of desalination, provision is made for intensive utilization of heat from the reactor, i.e., bringing it down to a low potential. Second, methods (both thermal and membrane types, especially the latter) for the desalination of salt water and methods of purifying and concentrating sewage, including liquid radioactive waste are based on the same technological processes. This generality is even more manifest in connection with the heightened requirements in regard to protection of the environment.

Papers presented at the symposium discussed in depth the experience gained from the operation of thermal desalination plants as well as the designing and construction of powerful new installations. The largest installations in operation include a multistage flash evaporation plant at Porto Torres, Sardinia, with an output of 52,800 m^3/day and a plant of the same type in Hong Kong (Japanese built) with an output of 180,000 m^3/day. A joint US –Egyptian paper made a comparative analysis of different sources of energy for desalinating water from the Red Sea and the Indian Ocean and came to the conclusion that it is desirable to use atomic energy for this purpose, notwithstanding the large oil reserves in the region.

As a whole, the work on thermal desalination of water has left the research and pilot-plant stage and is being pressed into commercial use.

Much attention is being paid to the effectiveness of the operation of thermal desalination plants: cutting capital outlays — especially metal consumption, increasing the specific yield of distillate, using low-potential heat, and constructing multipurpose installations and increasing their output. Some countries are drawing up comprehensive national programs for purification of natural saline, brackish water, and sewage.

Special attention is being paid to the development of membrane processes — electrodialysis, including processes at 65-70°C, and especially to reverse osmosis. Membrane processes for desalination of water are preferable to thermal processes and are therefore gradually displacing the latter as a result of lower energy consumption, lower metal consumption, and greater compactness.

The application of the reverse –osmosis method for treatment of sea, brackish, and discharged water, as well as the synthesis of reverse –osmosis membranes on the basis of cellulose acetate, new polymer materials, porous glass, etc. were the subject of about half of the papers, which testifies to the great interest in the subject. The Japanese company Kurita Water reported on the start-up of the world's largest reverse-osmosis installation with a output of 15,000 m^3/day. On January 1, 1975, the United States had 268 such installations with a total output of 150,000 m^3/day.

Greatest promise is held out by membranes in the form of so-called hollow fibers. At the present time, however, spiral modules with flat membranes constitute the principal type. The energy losses in the desalination of water by the reverse –osmosis method are 40% below those in distillation, the total expenses involved in producing fresh water being 20-40% below the cost of obtaining water in evaporators. It should be noted, however, that desalination by reverse osmosis requires careful preliminary treatment of the incoming water so as to prevent "poisoning" of membranes.

BOOK REVIEWS

V . I . Levin

NUCLEAR PHYSICS AND NUCLEAR REACTORS*

Reviewed by V . K . Vikulov

The book under consideration is a textbook for technical high schools.

The book is based on a course of lectures given by the author in the Obninsk Polytechnic School. The third edition of the book attests to its popularity, which results from its rather advanced scientific yet under-standable presentation of the subject.

The first part of the book, "Atomic and Nuclear Physics," comprises ten chapters.

Chapters 1 and 2 present information from atomic physics, the special theory of relativity, and quantum mechanics.

Chapters 3-5 deal with structure and characteristics of atomic nuclei, the properties of nuclear forces, and the main models of the nucleus.

In these chapters the reader is acquainted with radioactivity, radioactive materials and their utilization in the science and industry, the basic laws and types of radioactive decay, and the interaction of ionizing radiation with matter.

In Chapter 6 the most widely used instruments for recording ionizing radiation are considered. Chapter 7 describes the methods of accelerating charged particles and the designation and design of various reactors.

Chapter 8 is concerned with nuclear reactions. The author properly emphasizes the goal of research on nuclear reactions, energy and momentum conservation laws, and the probabilistic aspect of nuclear reactions. Particular attention is paid to thermonuclear reactions. The conditions required for a self-sustaining fusion reaction to take place in stars and on the Earth are formulated.

Chapter 9 briefly describes cosmic radiation and gives a classification of elementary particles.

Chapter 10 is a short description of both properties and sources of neutrons, of the principles of neutron spectrometry, and of the methods of recording neutrons. The interaction of neutrons with matter is considered in detail. The theory of nuclear fission and neutron multiplication is briefly outlined, but at this point some remarks should be made. Upon reading § 10.3, the reader has the impression that there exists only the spec-troscopy of slow neutrons (of less than 100 eV), which, of course, is not the case.

The setup, the basic scheme, and the classification of reactors, as well as the properties of reactor materials, are considered in Chapter 11 of the second part, "Nuclear Reactors." The classification of the reactors is incomplete. For example, heterogeneous reactors (see § 11.3) must be subdivided into tank and channel reactors, their advantages and shortcomings must be compared, and the basic possibility of recharging the fuel during operation of channel reactors should be specifically mentioned. High-temperature reactors have not been mentioned. It should have been stressed in § 11.6 that minimal noxious absorption of neutrons is the main requirement to be satisfied by all reactor materials.

Chapter 12 briefly describes the elementary theory of the criticality of thermal reactors, the neutron distributions in the various reactor types, and the importance of neutron leakage.

Chapter 13 explains the principles of reactor control, mentions the importance of delayed neutrons and of the temperature coefficient of the reactivity, briefly discusses the problems of burn-up and nuclear fuel regeneration and some details of switching-on and switching-off reactors, and gives an idea of the heat-exchange problems in reactors.

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* Textbook for Technical High Schools, 3rd edition, Atomizdat, Moscow (1975), 18, 34 l., 82 kopecks.

But formula (12.1) for the utilization coefficient of thermal neutrons (§ 12.1) is not correct, because the formula is excessively simplified without the required explanations. The Doppler effect and its importance for reactor control were disregarded.

Chapter 14 is a review of nuclear power generation in the USSR. The text mentions the capital expenditures and the cost of electric power generation by nuclear power plants, and describes the characteristics and design details of research reactors, experimental reactors, and reactors for power generation. It should be noted that some errors have been made in § 14.4. In particular, the possibility of replacing the graphite moderator by heavy water has not been provided in the design of the RBMK reactor; the fuel element shells of this reactor have a thickness of 0.9 rather than 0.45 mm; and outdated data were used to describe the Beloyarsk Nuclear Electric Power Station.

However, these shortcomings are immaterial and do not reduce the good overall impression which the book provides.

The main advantages of the book are its methodological uniqueness and its understandable presentation. It is important for a textbook that it includes data from the history of the respective fields of science and technology, examples for solving certain problems, and handbook data.

The book should be interesting and useful not only for students of technical high schools but also for students of colleges and for young specialists of other fields than engineering and physics.

R . M . Kogan, I . M . Nazarov, and Sh . D . Fridman

PRINCIPLES OF THE γ SPECTROMETRY

OF NATURAL MEDIA*

Reviewed by E . M . Filippov

The book is devoted to one of the important problems of nuclear geophysics. The utilization of γ spectrometry in daily work has greatly expanded the possibilities of radiometric investigations. Gamma spectrometry has made it possible to determine not only the distribution of the natural radioactive elements in natural beds but also the contamination of the earth's surface with the products of nuclear explosions.

The second edition of the book was prepared by authors who in our country were the first to use γ spectrometry for aerometric and autoradiometric surface surveying. In this section of radiometry they significantly contributed to the development of the theory, methodology, and applications.

The book comprises two sections with 11 chapters.

The physics of the γ spectrometry of natural media is considered in the first section which consists of five chapters. This part of the book provides information on the radioactivity of natural media according to the most recent data; natural and artificial radioactive isotopes and the radiation emitted by them are considered. Particular attention has been paid to describing the γ-radiation field in terms of mathematics. The principles of the theory of measuring γ fields, the determination of the source parameters of γ radiation and of the absorption of a medium from measured γ spectra, and the practical application of γ spectrometry are described in detail. The authors considered the sources of errors in measurements and data evaluation. Spectrometry with semiconductors is considered in addition to scintillation spectrometry.

The second section of the book relates to the γ spectrometry of natural formations (soils, deserts, and magmatic rocks) and describes specific examples. The authors describe the utilization of γ spectrometry in the search for sites of radioactive ore, oil-carrying beds, rare metals and nonferrous metals, and agronomical ore. Apart from this, the book tells of the use of γ spectrometry in investigations of the global

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*Atomizdat, Moscow (1976), 367 pp. 2 figures, 53 kopecks.

contamination of the natural environment by nuclear explosions, and in measurements of the water reserves in the snow cover and of the soil moisture with the aid of aerial surveying. The last chapter deals with the γ spectrometry of planets and meteorites.

However, when the authors describe the practice of γ spectrometry in nuclear-geophysics research performed in boreholes which were drilled in search and exploration work related to minerals, the authors restrict the considerations to some examples of uranium and thorium determinations. They have completely omitted the principles of the theory of spectral γ logging. In the last few years γ spectrometry has been employed in investigations of ocean-bottom sediments and in determinations of potassium in seawater in in situ measurements, of ^{137}Cs, and of nuclear-explosion products which have entered the seas and oceans. Hydrophysicists use the distribution of these elements in the water to draw conclusions concerning the transfer of water masses in the seas and oceans. These problems have been completely disregarded by the authors of the monograph.

These are very private remarks which shall not reduce the good overall impression provided by the book. The presentation is on a high scientific level. The book will be useful for a large number of specialists: scientists, candidates, and engineers. The book can also be used as a textbook for students who specialize in nuclear geophysics.

V. A. Bobrov, F. P. Krendelev, and A. M. Hofman

THE γ-SPECTROMETRIC ANALYSIS IN A CHAMBER WITH LOW BACKGROUND*

Reviewed by E. M. Filippov

This book deals with the improvement of γ-spectrometric analysis performed on various samples with low concentrations of natural radioactive elements which are important for solving geological, geochemical, and biological problems.

The book consists of an introduction, six chapters, and a list of references.

The first section refers to the γ radiation of the elements of the uranium and thorium series; the contribution of fluorescence radiation, measured with a germanium — lithium detector of $20.5-cm^3$ volume, to the total γ-radiation spectrum is estimated.

For the purpose of measuring low concentrations of rare elements in natural media, the Institute of Geology and Geophysics of the Siberian Division of the Academy of Sciences of the USSR has built a unique low-background chamber described in the second section of the book. The third section describes spectrometric measurements performed on various objects with the aid of single-, double-, and triple-crystal detectors.

The fourth and fifth sections are concerned with the method of measuring single-, double-, and triple-crystal spectra and the analysis of various factors influencing determinations made in sample of rare elements.

The sixth section pertains to determinations of rare elements in samples and provides information on the accuracy of measurements and on the sensitivity threshold of the equipment used.

When samples with a mass of 1.5 kg were measured in a single-crystal spectrometer with a 150×100-mm crystal (duration of a measurement 1 h), the following sensitivity thresholds, expressed in percent of the mass, were obtained: $7 \cdot 10^{-6}$ for uranium (determined via radium), $1.6 \cdot 10^{-5}$ for thorium, and 0.02 for potassium.

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*Transactions of the Institute of Geology and Geophysics of the Siberian Division, Academy of Sciences of the USSR, No. 329, Nauka, Novosibirsk (1975), 60 p., 37 kopecks.

Some remarks on the book must be made. The authors determine uranium in samples only via radium. But in natural objects the equilibrium between the rare elements can be disturbed; the book does not mention this point.

Unfortunately, spectrometry with semiconductors was disregarded, though the advantages of semiconductor instruments over scintillation detectors in measurements of low-energy radiation were mentioned in the first section. But the authors have shown that by measuring the 93-keV radiation one can determine ^{238}U in samples with a threshold sensitivity of $2 \cdot 10^{-5}\%$ [see Geologiya i Geofizika, No. 7, 118 (1971). Furthermore, semiconductor detectors can be used to determine uranium from the 63.3-keV γ radiation in samples [see At. Energ., 35, No. 5, 352 (1973)].

All these problems should have been treated or at least listed in the book.

Despite these shortcomings the book has been written on an advanced scientific level and will certainly be useful for specialists working on the analysis of natural and artificial radioactive elements in various samples.

A. S. Serdyukova and Yu. T. Kapitanov,

THE ISOTOPES OF RADON AND ITS DECAY PRODUCTS IN NATURE*

Reviewed by E. M. Filippov

The book is on the distribution of radon and its decay products in nature.

The book comprises nine chapters and a large bibliography.

In Chapters 1-3, the formation of gaseous products (emanations) in the radioactive decay of the elements of the uranium and thorium series, the physical and chemical properties of these products, their spread and transfer in nature, and the existing forms and concentrations in various media are considered.

Chapter 4 gives data on the radon concentration in living organisms and describes the biological effects of radiations and radiation safety standards.

Chapters 5-7 describe methods of determining the emanating coefficients and the amounts of emanations from rocks, water, and air. The dosimetric monitoring of individuals is considered too.

Chapter 8 describes measures for protecting personnel working under conditions of increased concentrations of emanations and their decay products in air.

Chapter 9 describes the utilization of emanation — measuring methods for the detection of sites with radioactive or nonradioactive minerals, for geological mapping, and for research on atmospheric phenomena and in medicine.

The biological effects of ionizing radiation, the radiation safety standards (Chapter 4), the dosimetric monitoring of individuals (Chapter 7), and the measures for protecting personnel working under conditions of increased concentrations of emanations and their decay products (Chapter 8) should be treated at the end of the book. The book would benefit from such a rearrangement.

The authors described in detail the determination of radon in friable geological deposits and the use of the emanation method for solving geological problems. The book should therefore include a description of the field emanation meters which are supplied by the Russian industry. The book then would give the impression of greater comprehensiveness. But these shortcomings concern only particular aspects.

The book has been written on an advanced scientific level and will be useful for a large number of readers — geophysicists, geologists, and other specialists. The book can serve as a textbook for post-graduate students and students specializing in the application of radiometric techniques to searching for several solid minerals.

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*2nd edition, Atomizdat, Moscow (1975), 16, 15 l., 1 rouble 78 kopecks.

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